September 1958

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NUCLEONICS

ATOMIC POWER • NUCLEAR ENGINEERING • APPLIED RADIATION

Photographed at Geneva, September 1, 1958

FUSION, FISSION, RADIATION:

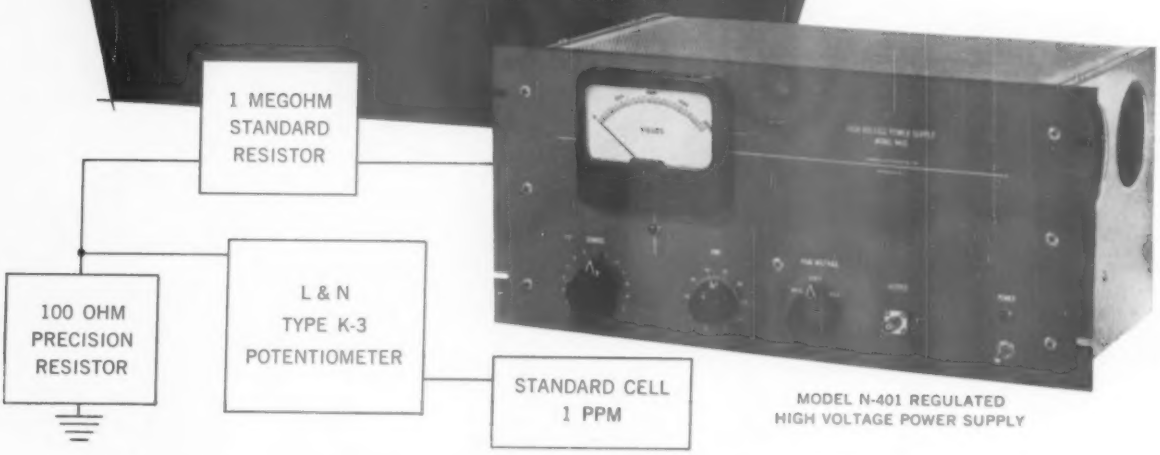
NUCLEONICS Reports on Revelations at U N Conference

GENEVA—1958

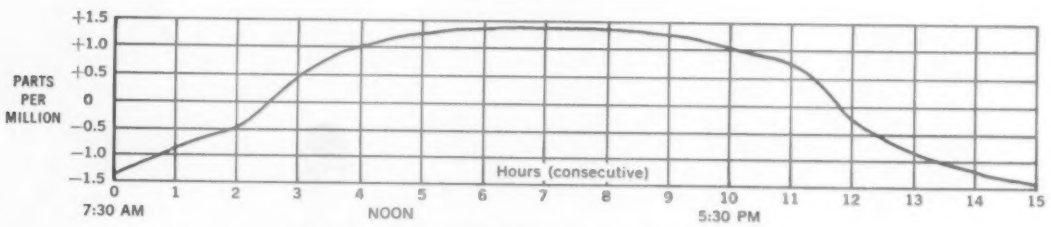
Hammer

POWER SUPPLY SURPASSES EXPECTATIONS

The Ultimate in Stability • Proved In Precision X-Ray Spectroscopy Laboratory Test



MODEL N-401 REGULATED HIGH VOLTAGE POWER SUPPLY



Graph Shows Long Term Stability To Be Better Than ±1.5 Parts Per Million

The Johns Hopkins University Precision X-Ray Laboratory recently tested the stability of a Model N-401 Regulated High Voltage Power Supply *for a period of 15 hours. The voltage was constant within ± 1.5 part per million* as shown by the graph. The power supply was in use approximately 2000 hours and tested in an ambient temperature of 24.00° ±0.05° centigrade.

Engineering Department expanding.
Positions open for experienced men.

- **The Model N-401 is a new concept in regulated power supply design — it provides up to 5 MA of current and negative or positive DC voltage — continuously variable from 500 to 1800 volts.**
- **Unmatched stability against line change; less than 0.003% per volt — Regulation, 0.0025% per MA @ 1000 volts — Output impedance, 2.5 ohms.**
- **Low noise and ripple; less than 1 MV RMS.**
- **Designed for use with scintillation and proportional counters or for general laboratory use — anywhere the ultimate in stability is required at a reasonable price.**

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P. O. Box 531, Princeton, N. J.,
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NUCLEONICS
ATOMIC POWER
NUCLEAR ENGINEERING
APPLIED RADIATION

VOLUME 16 NUMBER 9

SEPTEMBER 1958

Geneva-1958

A NUCLEONICS Special Report

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About NUCLEONICS

Papers, Papers, Papers

As the photograph below, and indeed all of this issue, testifies, it takes a lot of papers to make a Geneva Conference. For Geneva-1958 there were more than 2,400 finally accepted. The editors of NUCLEONICS had to scan each of these papers, at least by title, to come up with the set of 228 particularly important papers reported on in this issue. On one hand, it was a heck of a chore and we were glad when it was over. On the other hand, it was very stimulating—the whole world of nuclear technology paraded before us in a kind of sneak preview of the Conference which, as this was being written (Aug. 18), had not yet taken place.

PRETTY GIRL stands alongside stack of over 2,400 Geneva Conference papers studied for this special issue. Proceedings of the 1955 Conference are on desk at right. (Pretty girl is Dorothy Pappachristou, secretary to Chief Editor Jerry Luntz)

You see, our work began in mid-July, when the United Nations made the first of the Geneva papers available to us, embargoed for release during the Conference under an arrangement available to the technical press generally. There followed the editorial work of digesting the papers that seemed significant enough to report on in the 62 pages that could be made available by giving the entire September issue over to Geneva.

Some of the papers, the Russian papers in particular, were late in coming in, so we were still writing pieces of the story as late as August 25. The mop-up was left to the three-man team —Chief Editor Jerry Luntz, News Editor John Kenton and Associate Editor Hoby Ellis—that flew to Geneva for on-the-scene reporting.

All 12 pages of the Roundup section this month are devoted to that on-the-scene report, cabled from Geneva as late as September 11. Fast work all along the line and closing the form at the printer made it possible to include that up-to-the-minute account of the Conference itself and its personalities, color and significance.

Cover. "What about the cover?" you may ask. "That's from Geneva-1955, isn't it?" Fact is, it is not. We arranged to have a photograph of the opening session taken on September 1st —the opening day of the Conference— and air-expressed to New York. Quick work at the engravers enabled it to be fit into a specially designed cover plate and made ready for press within 48 hours.

All in all it has been great fun for us. We hope it proves valuable reading for you. There is a tremendous store of nuclear knowledge in the Geneva papers and we hope that in some small measure we have succeeded in reflecting it in the present issue. Inevitably, however, particular specialized work has had to be bypassed. For this we trust that the readers will turn to the definitive Proceedings of the Conference to be published early next year.

Appreciation. This special report on Geneva-1958 would not have been possible without the assistance of many people in atomic energy organizations here and abroad (who supplied originals of photographs when required), the McGraw-Hill correspondents in London and Paris (who helped expedite much valuable material), and, above all, the staff of the United Nations and its Geneva Conference Secretariat (who, from Conference Secretary-General Sigvard Eklund on down, helped in many ways). Our thanks to them all.

—The Editors

Coming Features in NUCLEONICS

Next Month

Patents in Atomic Energy
Reactor-Thermal-Design Criteria
Vapor-Gas Proportional Counter
Improved Reactor-Kinetics Constants

Later Months

Pulse Shaping in Nuclear Spectrometry
Shielding the Enrico Fermi Fast Breeder Reactor
Well Logging—A Review

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Assembling a test core for the Nuclear Merchant Ship Reactor project, preparatory to conducting critical experiments at B&W's Critical Experiment Laboratory, Lynchburg, Virginia.

B&W PROGRESS REPORT:

N.S. SAVANNAH—*Critical Experiments Underway*

The reactor core for the world's first nuclear merchant ship, the N.S. Savannah, is now undergoing final design at Babcock & Wilcox. A test core went critical on February 6, 1958. Experiments are now being run to verify core design calculations of such factors as flux distribution, reactivity and optimum fuel arrangement.

One of B&W's primary objectives is to achieve the longest core life possible with present technology. Present estimates are that a single loading of nuclear fuel will allow the Savannah to steam for 350,000 nautical miles—about 3½ years—based on a lifetime of 600 full power days.

The core design embodies a number of superior features applicable to marine propulsion. For example, fuel elements will contain slightly enriched uranium oxide, which is less subject to distortion under irradiation than is metallic fuel. Since the oxide is almost completely inert to water, hazards due to chemical actions between water and fuel, in the event of fuel element rupture, are reduced. Another significant feature is that the U-238 contained in this fuel will be converted to plutonium, another fissionable fuel, which will supplement the initial U-235.

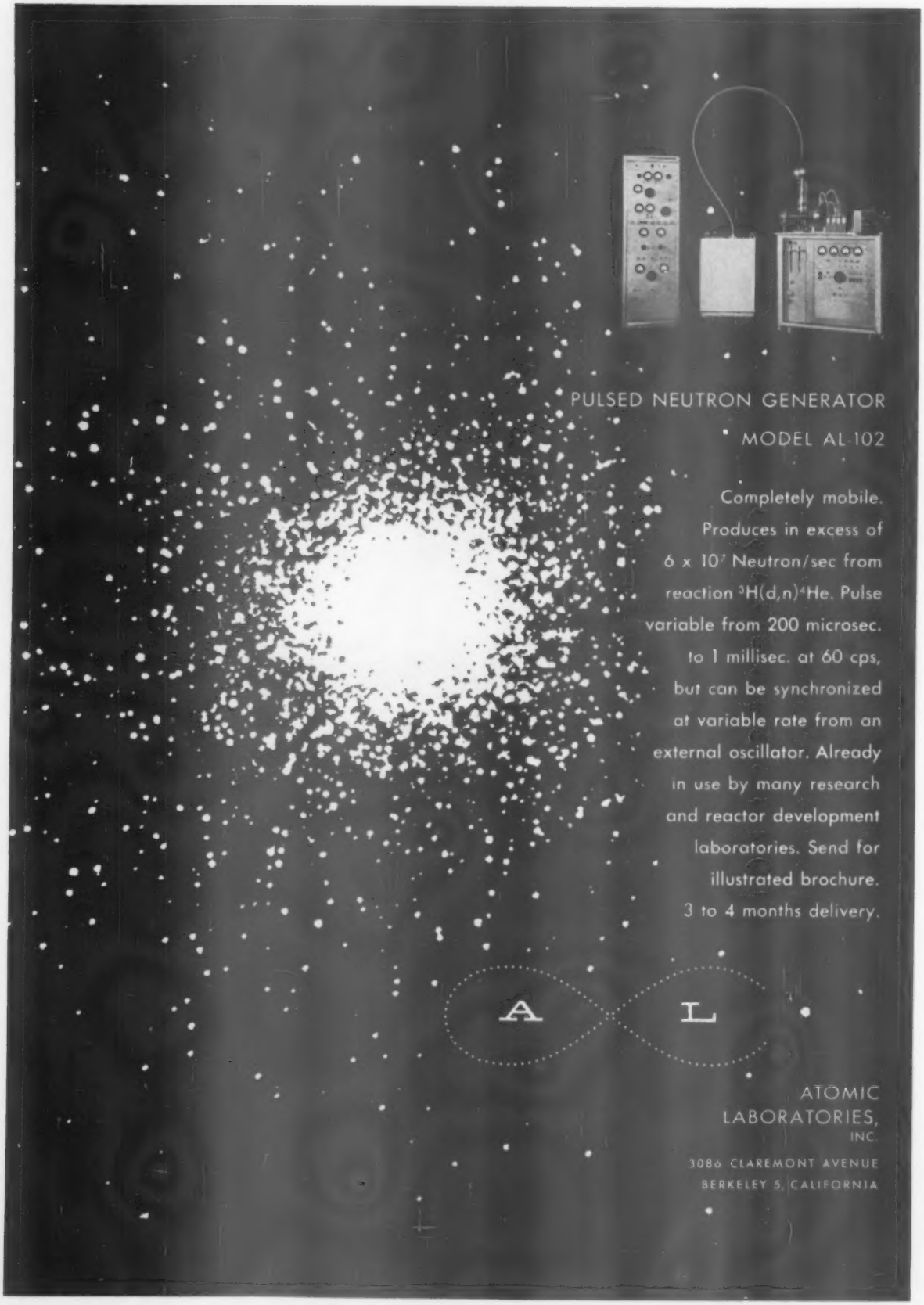
Concurrent with this series of critical experiments, the reactor vessel, heat exchangers, and pressurizer for the Savannah are being fabricated in B&W's heavy-component facilities.

The Babcock & Wilcox Company, Atomic Energy Division, 161 East 42nd Street, New York 17, N. Y.

AE-58

BABCOCK & WILCOX

ATOMIC ENERGY DIVISION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Completely mobile. Produces in excess of 6 x 10⁷ Neutron/sec from reaction ³H(d,n)⁴He. Pulse variable from 200 microsec. to 1 millisec. at 60 cps, but can be synchronized at variable rate from an external oscillator. Already in use by many research and reactor development laboratories. Send for illustrated brochure. 3 to 4 months delivery.
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TYPICAL CORROSION RESISTANCES OF ZIRCONIUM AND TITANIUM

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	ZIRCONIUM	TITANIUM
Sulfuric Acid	excellent to good below 80%	good below 5%
Nitric Acid	excellent	excellent
Hydrochloric Acid	excellent	good below 10%
Phosphoric Acid	excellent to fair below 85%	poor
Chromic Acid	excellent	excellent to good
Aqua Regia	poor	excellent
Wet Chlorine Gas	poor	excellent
Chlorine Water	excellent	excellent
Sodium Hydroxide	good below 90%	good below 50%
Ferric Chloride	poor	excellent
Calcium Chloride	excellent	excellent
Cupric Chloride	poor	excellent
Sodium Chloride	excellent	excellent
Ammonium Chloride	excellent	excellent
Aluminum Chloride	excellent	excellent to fair

Chart shows corrosion resistances of zirconium and titanium to typical chemicals.

Steam jet made of zirconium, which has given trouble-free performance after a year in hydrochloric acid service. For comparison, a throat piece from a steam jet (below) is shown after only a week of similar service.

By specifying titanium or zirconium for processing equipment, you can now overcome most of the corrosive media which attack other metals.

Even with such hard-to-handle chemicals as chlorides and oxidizing acids, equipment can have extremely long service life when made from these corrosion-resistant materials. Problems of product contamination in chemical and food processing can also be virtually eliminated.

Mallory-Sharon is in position to offer you *both* titanium and zirconium mill products for equipment fabrication—plus engineering assistance and unbiased recommendations on the most suitable material.

Titanium is now available from stock in a complete range of mill products, may be readily fabricated, and more than pays its extra cost where ordinary metals fail. Zirconium facilities are being rapidly increased, and mill shapes are now in production.

For information on the corrosion-resistant properties of titanium or zirconium write Mallory-Sharon Metals Corporation, Niles, Ohio.

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How nickel alloy equipment resists

Fluoride Volatility Process

To prevent attack by the coolant medium on fuel materials and to prevent escape of fission products, fuel elements are often clad with zirconium. To recover the uranium from spent fuel elements of this type, a pilot plant has been built at Oak Ridge National Laboratory, utilizing the following process:

1. Dissolution of the elements with anhydrous hydrogen fluoride in a sodium fluoride-zirconium tetrafluoride melt at 650°C :

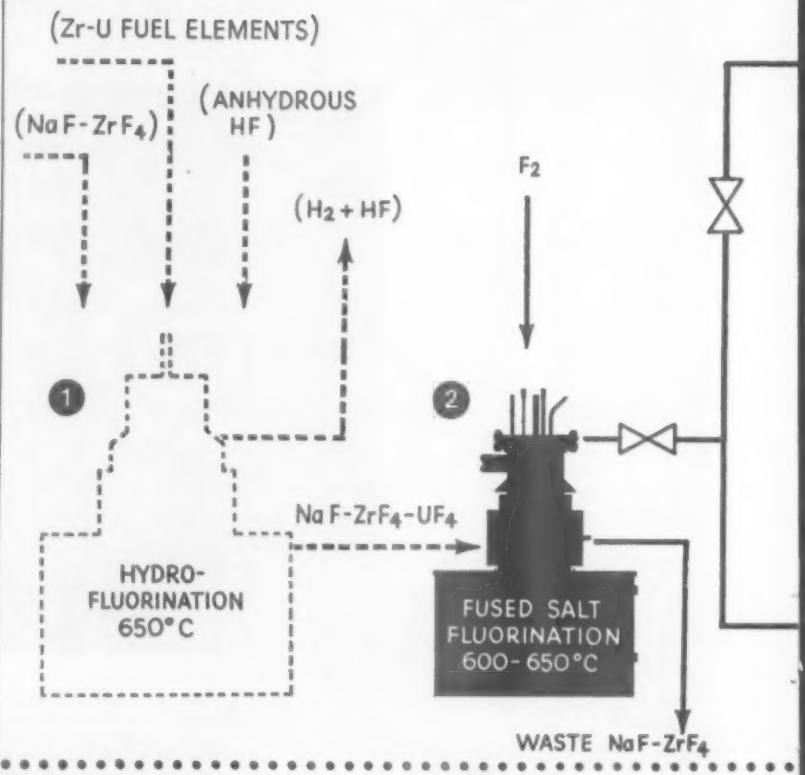
$$4HF + Zr \xrightarrow[650°C]{NaF\text{-}ZrF_4} ZrF_4 + 2H_2 \uparrow \quad (1)$$

$$4HF + U \xrightarrow{} UF_4 + 2H_2 \uparrow \quad (2)$$

2. Formation and volatilization of uranium hexafluoride from molten salt by contact with excess fluorine at 600°C :

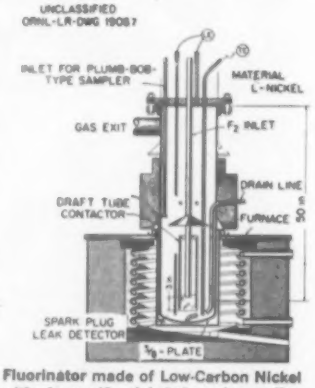
$$UF_4 + F_2 \xrightarrow[600°C]{NaF\text{-}ZrF_4} UF_6 \uparrow \quad (3)$$

3. Further purification of the uranium hexafluoride by absorption on sodium fluoride at 100°C , followed by desorption at 100° to 400°C with fluorine.
4. Collection of decontaminated uranium hexafluoride in cold traps.



Equipment

The high-nickel alloys used in the process equipment were selected because they provide a combination of corrosion resistance, heat conductivity, high-temperature strength and good electrical characteristics.



Fluorinator made of Low-Carbon Nickel with Inconel* nickel-chromium alloy liner.

Materials Evaluation

Oak Ridge reports after one year experimental operation . . .

Inconel alloy liner excellent. Low-Carbon Nickel shows corrosion, but is best material presently available.

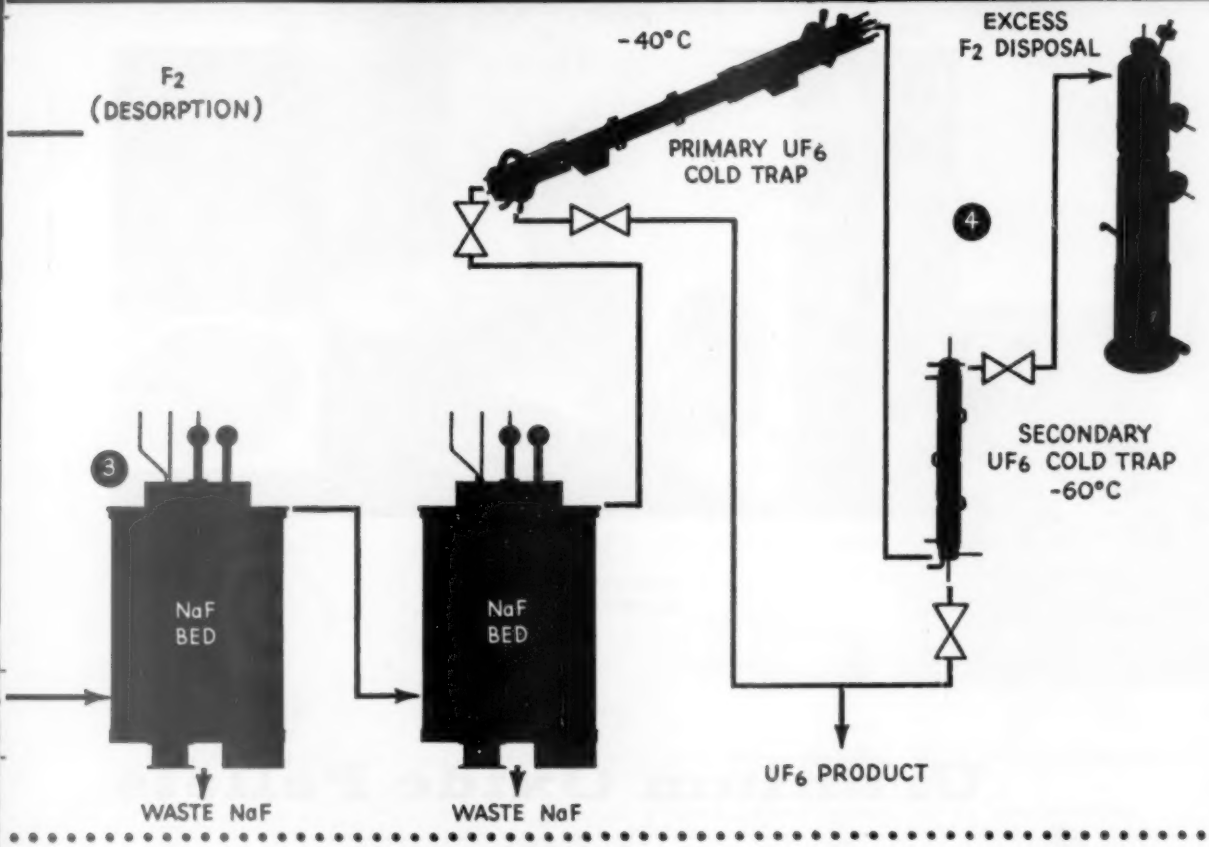
Literature

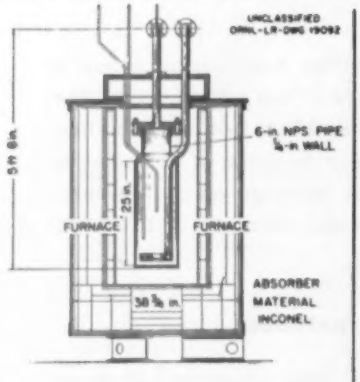
New 24-page booklet "Handling Fluorine and Fluorine Compounds with Inco Nickel Alloys" summarizes recent corrosion data, gives case histories on methods of handling. Write:

THE INTERNATIONAL NICKEL COMPANY, INC.
67 Wall Street
New York 5, N. Y.

*Registered trademark

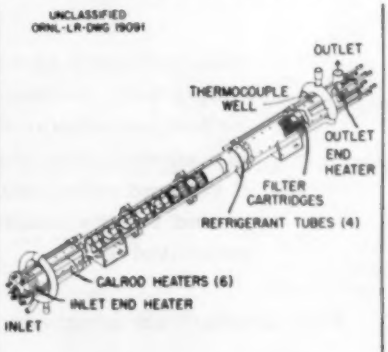
corrosion in uranium recovery process





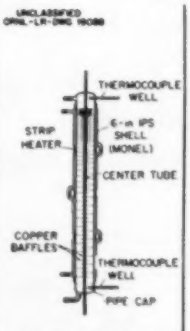
Absorber entirely of Inconel alloy.

Evaluation: Good results with Inconel alloy.



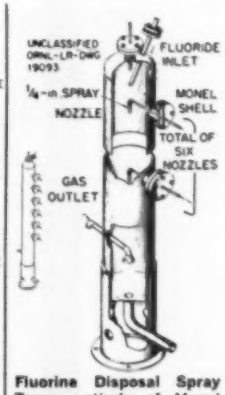
Primary Cold Trap — one end packed with Nickel wire mesh filter cartridges for 12 inches.

Evaluation: Undetermined; no trouble expected.



Secondary Cold Trap — Monel* nickel-copper alloy shell, gasket, pipe connections.

Evaluation: Undetermined; no trouble expected.

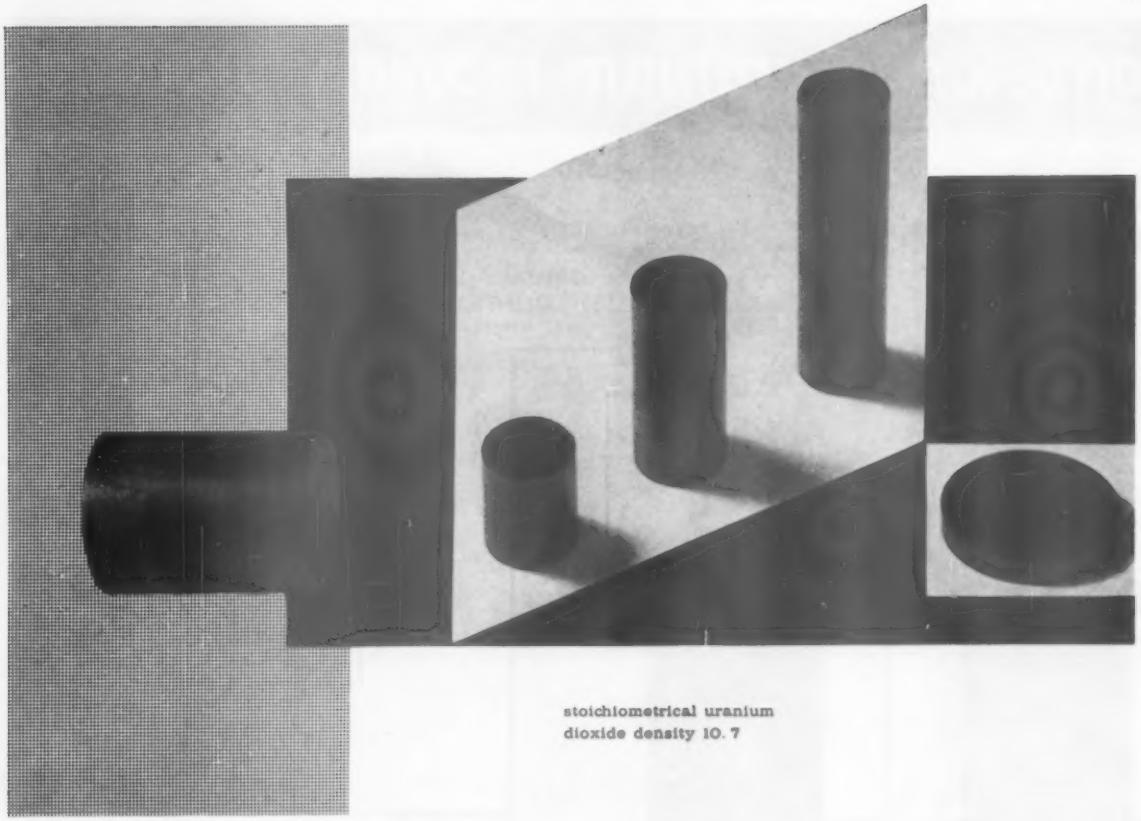


Fluorine Disposal Spray Tower entirely of Monel alloy.

Evaluation: Good. Only evidence of corrosion at point of reaction of KOH and fluorine at nozzle . . . was expected.

(Information courtesy of Oak Ridge National Laboratory, operated by Union Carbide Corp. for the U.S. Atomic Energy Comm. Adapted from *Industrial & Engineering Chemistry*, Feb. 1958, by permission of copyright owner.)

INCO NICKEL ALLOYS



stoichiometrical uranium
dioxide density 10. 7

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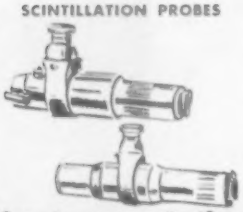

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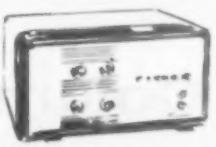
SPECTROMETERS



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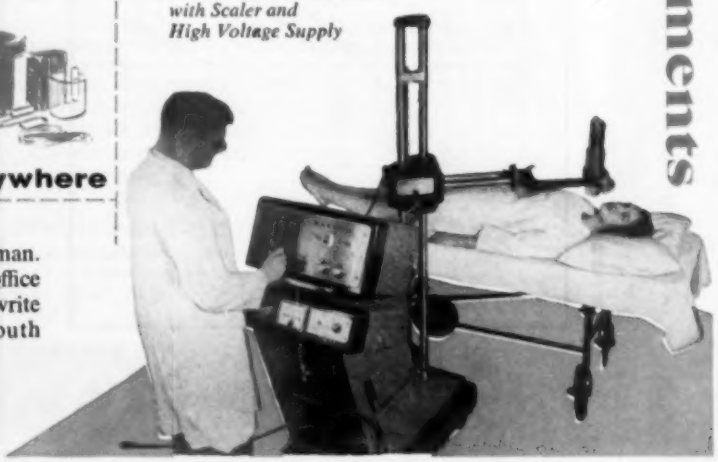
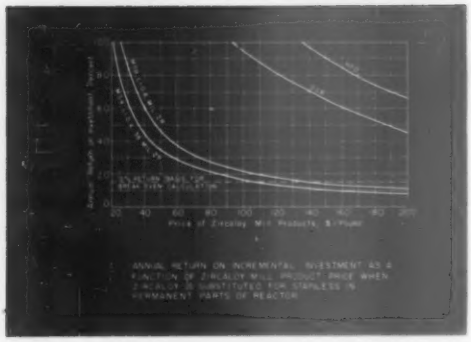
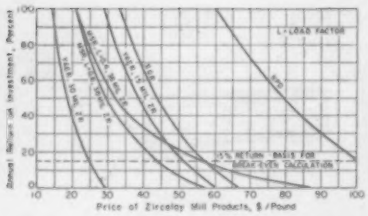


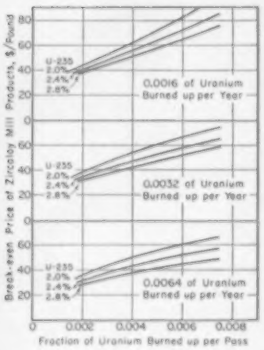
PICKER nuclear instruments



ANNUAL RETURN ON INCREMENTAL INVESTMENT AS A FUNCTION OF ZIRCALOY MILL PRODUCT PRICE WHEN ZIRCALOY IS SUBSTITUTED FOR STAINLESS IN PERMANENT PARTS OF REACTOR


L = LOAD FACTOR
15% RETURN BASIS FOR BREAK-EVEN CALCULATION
Annual Return on Investment, Percent
Price of Zircaloy Mill Products, $ / Pound

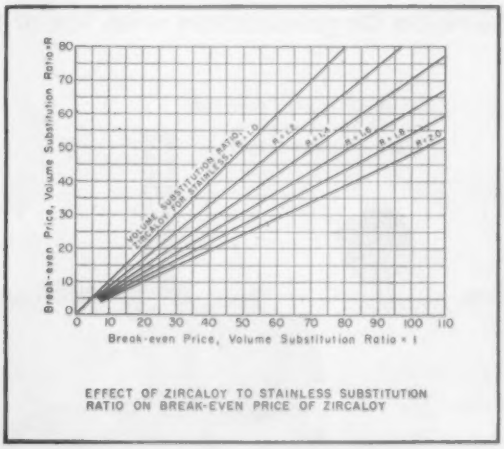
ANNUAL RETURN ON INCREMENTAL INVESTMENT AS A FUNCTION OF ZIRCALOY MILL PRODUCT PRICE WHEN ZIRCALOY IS SUBSTITUTED FOR STAINLESS AS FUEL CLADDING


U-235
2.0%
2.4%
2.8%
0.0016 of Uranium Burned up per Year
0.0032 of Uranium Burned up per Year
0.0064 of Uranium Burned up per Year
Break-even Price of Zircaloy Mill Products, $/Pound
Fraction of Uranium Burned up per Pass

EFFECT OF VARIATION OF FUEL ENRICHMENT, BURNUP FRACTION AND RATE OF BURNUP.

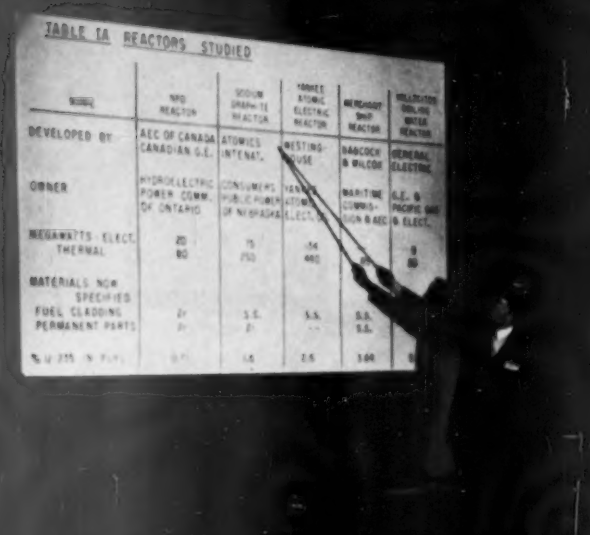
EFFECT OF LOAD FACTOR ON BREAK-EVEN PRICE OF ZIRCALOY MILL PRODUCTS IN MERCHANT SHIP REACTOR.

Columbia-National Study Shows...

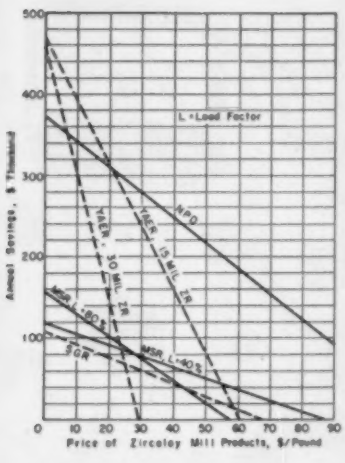


EFFECT OF ZIRCALOY TO STAINLESS SUBSTITUTION RATIO ON BREAK-EVEN PRICE OF ZIRCALOY

NEW METALS FOR INDUSTRY



Dr. Manson Benedict, head of the Nuclear Engineering Department at M.I.T., points to data on the reactors studied, at a recent symposium held to present the results to industry representatives.



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A summary of the results of this study will be sent to those requesting it on business letterheads.

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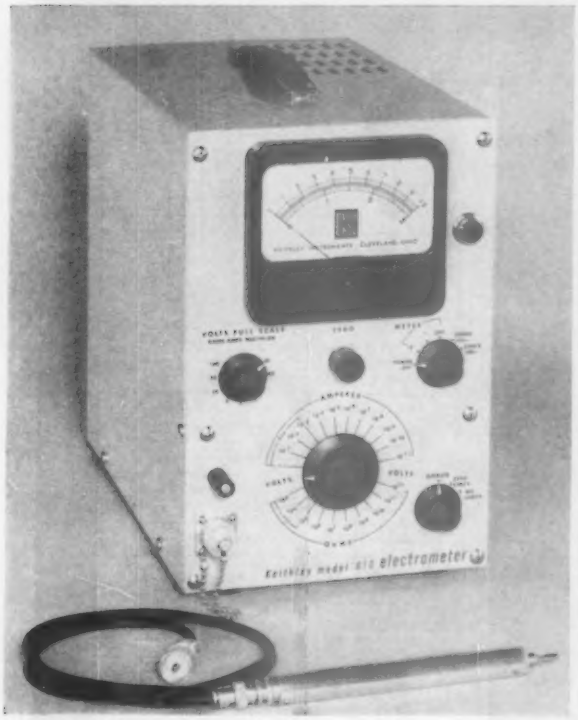

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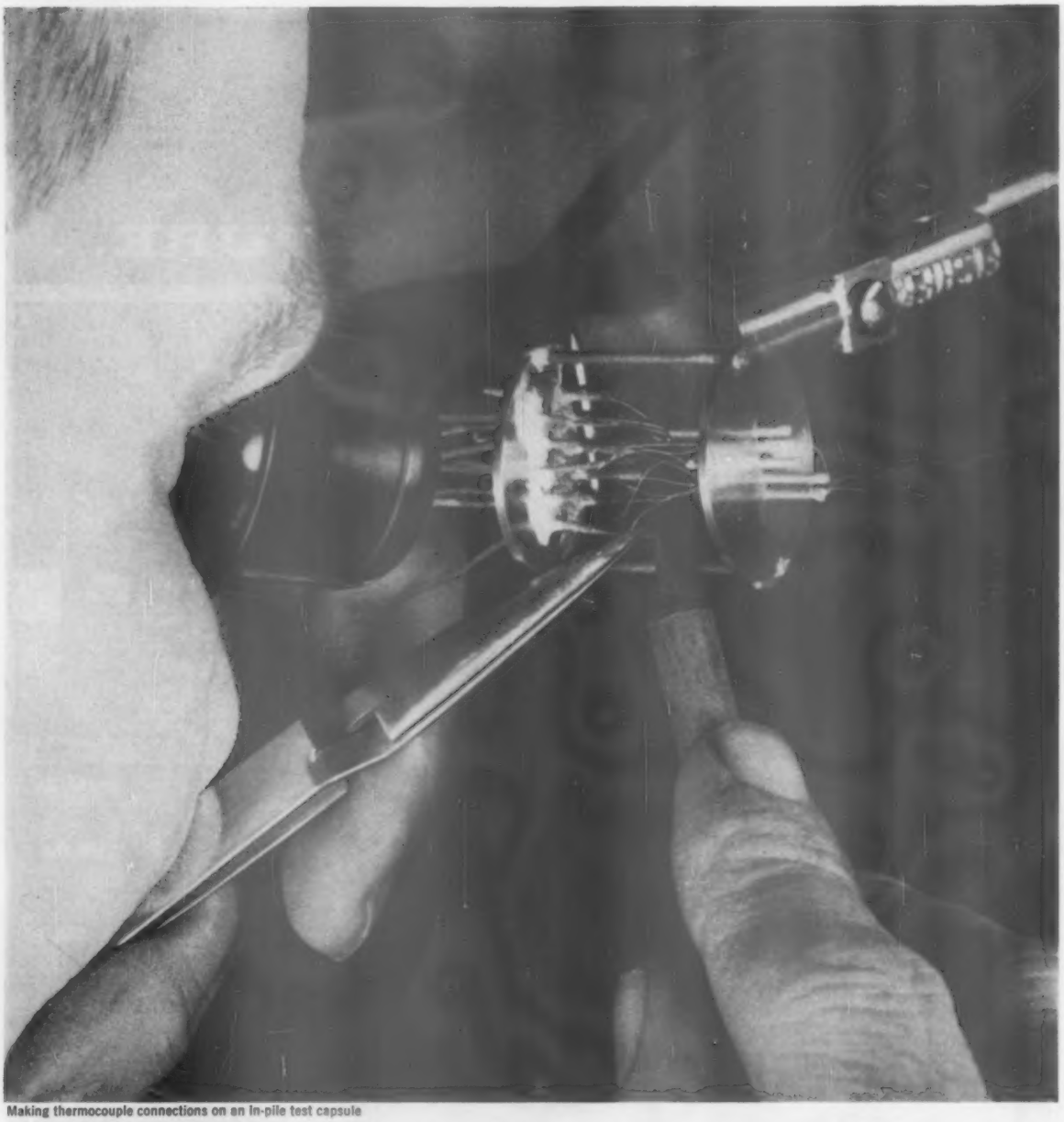

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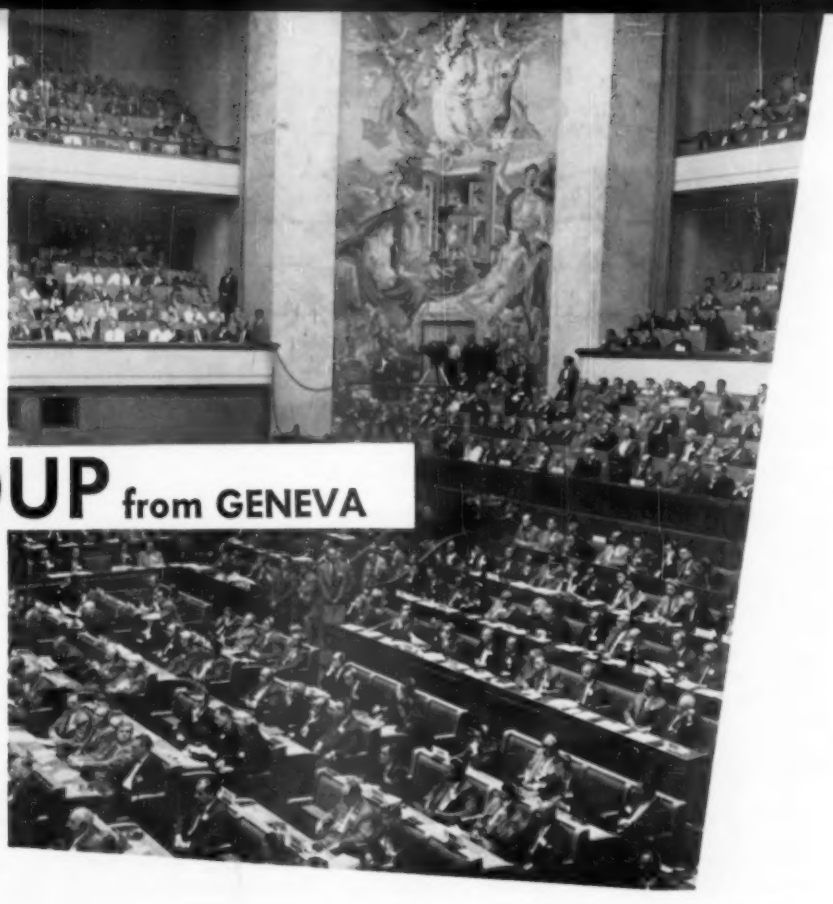
ROUNDUP from GENEVA

September 1-13



NUCLEONICS' On-the-scene Report On the 1958 Geneva Conference

Geneva, Switzerland, was the center of the nuclear world for two weeks this month as the United Nations held its Second International Conference on the Peaceful Uses of Atomic Energy. More than 5,000 delegates and observers from 69 countries attended technical sessions, a scientific exhibit and a commercial exhibit.

Although there was electricity in the air as the crowds converged on the main assembly hall for the opening plenary session, there was not the same excitement as there had been at the first conference in 1955. Attendance quickly dropped through succeeding sessions until the opening of the first fusion discussion, which commanded considerable attention.

Controlled thermonuclear power then held the spotlight for the remainder of the conference—in the sessions, in the exhibits and in informal discussions. Great emphasis was placed on the need for international cooperation in fusion.

Despite the clear pinpointing of the relative timetables of fission and fusion at the conference, fission took a back seat most of the time.

This issue of NUCLEONICS, starting on p. 61, features a report on the papers presented at Geneva. This 12-page Roundup report cabled from Geneva, is based on coverage of the actual meetings and exhibits by a three-man NUCLEONICS team.

Corridors were more crowded than sessions.

Geneva Disappoints Many Attendees

GENEVA, *September 1–13*

In August, 1955, the Palais des Nations in Geneva was the scene of what was probably the world's most successful and significant scientific conference—the First United Nations International Conference on the Peaceful Uses of Atomic Energy. East and West, for the first time, exchanged information on matters that had been secret and learned to respect each other's competence. Small nations got their first comprehensive picture of the state of nuclear technology. And international vendors and buyers of nuclear equipment were able to meet for the first time in the same marketplace.

These were some of the accomplishments of Geneva-'55. They represented the target at which Geneva-'58 had to shoot, for inevitably any evaluation of '58 must be done in comparison with '55.

It is now clear (as this is being written, towards the end of the conference) that Geneva-'58 fell far short of the mark. Although there were several noteworthy aspects of the conference, the question that nagged many observers at the conference was, "Was it worth all the money and herculean effort that went into it?"

Credit Side

The clear-cut accomplishments of Geneva-'58 were:

1. Controlled thermonuclear power is now totally declassified throughout the world.

2. Full international interchange of peaceful nuclear technology now seems much more possible.

3. A clearer understanding of the problems of and time scale for achieving successful controlled thermonuclear devices emerged.

4. Despite remaining gaps in information, the Soviet fission program is easier to understand.

5. The American exhibit had considerable impact on the world nuclear community.

Debit Side

The shortcomings of Geneva-'58 were:

1. Very little information of "earth-shaking" significance was presented.

2. A not much better picture of nuclear power economics came out than was presented in '55.

3. In certain fields, particularly fusion, many workers were kept from their regular scientific duties for many months because of Geneva preparations.

Over-all View

Because an overriding purpose of the conference was political—as it was in '55—it lacked the flavor of a scientific meeting that so many were looking for. If nothing else, the mammoth nature of the entire affair tended to overwhelm people, physically and mentally.

Thus, many participants were disappointed and accepted their roles merely as a "duty" for their governments.

For many, the one interesting thing at the conference was the American scientific exhibit (*see p. 20*). By all odds, the exhibit was one of the most impressive displays on nuclear energy ever set up. It certainly far surpassed, in quality and quantity, the exhibits of every other country showing at the Palais des Nations.

Whereas in 1955 the USSR exhibit ranked first on curiosity interest, this time the American presentation was the main attraction, with the fusion sector in the center of the stage.

Fusion. At Geneva-'55, the president of the conference, Homi J. Bhabha of India, alerted the world to the potential of controlled thermonuclear power.

At Harwell in 1956, the USSR's Igor Kurchatov gave the world the first scientific discussion of fusion work. (The Soviets took and were given, at Geneva-'58, ample credit for this first declassification.)

At Geneva-'58, total declassification of fusion work was announced. This, to many, made the whole conference worthwhile.

This was pointed up very well in the leading Russian fusion paper by L. A. Artsimovich: "For the first time, these results will be discussed on an international scale, and this is probably the most important step that has been made towards the solution of this problem. The importance of this fact is greater than that of the separate investigations, which as yet have not brought us very much nearer to our ultimate goal." He then called for international cooperation to assure success.

The fact that declassification of fusion would be an important accomplishment of the conference was made clear at the outset when the US and UK, on the eve of the conference, dramatically announced "total declassification." Ideas in the controlled thermonuclear field will no longer be "born classified", according to the US. One effect of this within the US will presumably be the abolition of the need for private companies to get "access permits" for access to government fusion data.

The USSR statement on its declassification of fusion came out only in answers to questions at press conferences rather than in any formal way, and then, perhaps because of lack of clarity of communications, the answers came out begrudgingly. The main evidence of declassification was stated to be a set of four

volumes, covering the whole Soviet fusion program, that was presented to the UN by chief delegate V. S. Emelyanov immediately after he finished his address on opening day.

One of the great benefits of the declassification of fusion came out right at the conference. With secrecy removed, it was possible for the delegates to go away with a realistic appraisal of the true state of the fusion art, to know that, despite all the clamor about the subject, its full realization was still quite some years off (*see p. 22*). The delegates could be certain that they had gotten a total picture since fusion researchers had interrupted their work en masse, to take stock of it in the papers and exhibits.

A byproduct of the fusion presentations was the debunking of one landmark of fusion success and the pointing up of another. Artsimovich said that whether or not fusion neutrons are thermonuclear in origin is of more interest today to newsmen than it should be to scientists. Sir John Cockcroft of the UK seemingly echoed this when he defined a fusion "breakthrough" as occurring when the power output from a fusion device is at least equal to that which is put into it.

Soviet program. As in '55, one of the most eagerly awaited parts of the conference was the presentation of Soviet results. In fusion, virtually nothing beyond the Kurchatov talk had come out in the past several years. In fission, a paper at the Belgrade World Power Conference in 1957 (NU, July '57, p. 20) had been the only important release.

But at Geneva-'58, the world got a good look at both the Russian fusion and fission work (*see pp. 23 and 25*). Interestingly, both seem to have as wide a scope as that of the US, if not the same depth.

The lack of need of the USSR for a new source of power, which has been suspected, was documented by Emelyanov: "Our country possesses tremendous reserves of organic fuel of all types as well as considerable resources of water energy."

Nevertheless, the USSR was able to pull off an effective coup midway in the Conference when it announced full-power operation of its 100 Mw(e) reactor in Siberia. A major effect of the disclosure was to raise doubts about the Western assumption that Russia has little in the way of a reactor construction program.

Nuclear power economics, experience. One of the biggest disappointments of the conference was that, although much experience has been had in reactor technology since Geneva-'55, there has not been enough to make nuclear power plant costs much more predictable than in '55.

For example, although the UK has been talking about getting a fuel irradiation of 3,000 megawatt-days per ton in Calder-type reactors, the maximum it has achieved thus far has been 1,300 Mwd/ton in "hundreds of elements in certain channels" of the No. 2 reactor at Calder.

And, although US firms have been offering up to 10,000 Mwd/ton, no such experience in full fuel elements was reported at Geneva—this despite the fact that the US has now had 17 reactor years of operation compared with only 7 at the time of Geneva-'55.

But despite this lack of useful cost data, Cockcroft predicted that the world's present nuclear power plant capacity of 200,000 kw would be upped to 15,000,000 kw by 1970.

A supporting note was added in a World Bank-Italian report on the SENN project. Bids from vendors fell in a range of 11.1 mills/kwh to 16.9 at 80% plant factor and capital charges of 14%. Apparently, the world nuclear industry is now at the point where it is ready to risk its reputation on providing low-cost nuclear power plants.

Plowshare. One sour note at Geneva came in connection with the US paper on Project Plowshare (peaceful applications of nuclear explosions) and resulted in the only open bitterness between the US and the USSR. Because the US speaker, Gerald Johnson of Livermore, gave the only direct discussion of weapons on the program, he came in for sharp questioning from Japanese delegates and the UK, and for direct criticism from the USSR. Behind the conflict were: a general sensitivity on weapons; a feeling that Plowshare might be a means to get around a ban on weapons testing; and a belief that Plowshare might represent a device to keep some nonmilitary thermonuclear data classified. It was the opinion of many that the US didn't come off well in this presentation.

Fusion leaders enjoy declassification

Three "stars" of Geneva-'58 after reading their fusion papers (from left): E. I. Dobrokhotov (who read the USSR paper for L. A. Artsimovich), P. C. Thonemann, UK; and Edward Teller, U. S.

New Chairman tours exhibits

Chairman John A. McCone (striped tie) of USAEC being shown through U. S. exhibit by U. M. Staebler of AEC's reactor development division

U. S. COMMERCIAL EXHIBITS; U. K. exhibits are at left

ARGONAUT construction started Monday morning, Sept. 1; reactor went critical before end of week

OPERATING REACTORS in commercial exhibits—at left, Atomics International water boiler; at right, Aerojet-General AGN-201

"Phenomenal" Is the Word

ROTATING-PLASMA device fired discharges and 'scopes plotted data

GENEVA, *September 1–13*

The best word for the official American exhibit at Geneva is "phenomenal"! Even blasé Americans were overwhelmed by it, in addition to many of the foreign scientists.

The exhibit was so successful for many reasons: new information was presented, many actual working devices were in operation, experiments were carried out on the spot, there was an interrelationship between many of the experiments, the top experimenters for each of the projects represented were present themselves and the whole thing was very well organized.

Although the very elaborate fusion display stole the show, there was much else that was interesting:

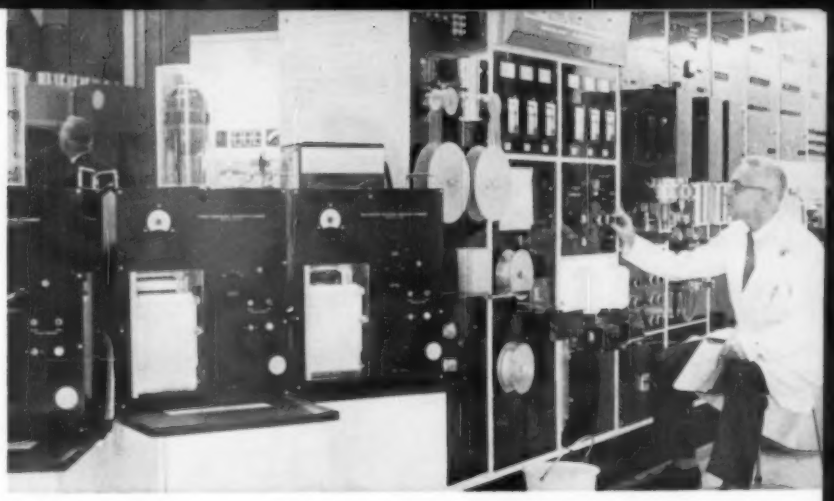
1. An operating reactor (TRIGA) produced radioisotopes that were used in various experiments in the American exhibit.

2. A health and safety laboratory kept watch over the entire US exhibit via remote monitors to maintain a maximum of 2 mr/hr at any viewing point in the show. Radioactive samples were counted throughout each day.

3. An Argonaut reactor was constructed at the exhibit during the first week of the conference to demonstrate the ease of construction.

4. A computer center provided services for nearly a dozen different experiments at the US exhibit.

HEALTH PHYSICS was not ignored— Hanson Blatz of AEC New York Operations office at work at health and safety exhibit, which monitored all U. S. exhibit radiation facilities



STABILIZED PINCH device exhibited by University of California Radiation Laboratory included high-speed camera to photograph discharges

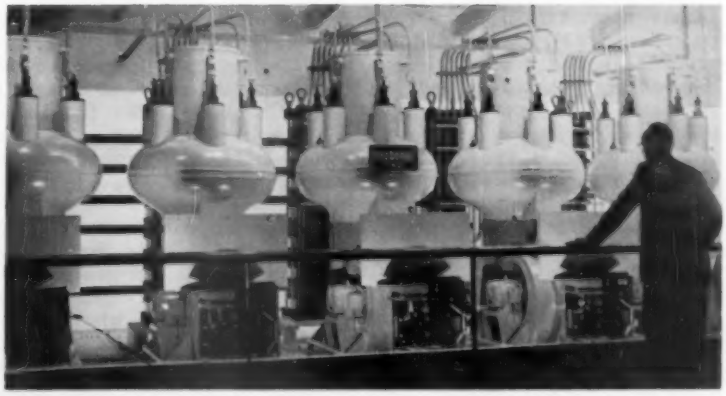
TRIGA reactor, in operation in U. S. exhibit, produced radioisotopes for use in exhibit experiments

for United States Exhibit

These are only a very few of the exhibits. The visitor to these and the others could easily come away with the impression that he was in one of the national laboratories in the US.

Confirming that impression would be a visit to the basement of the US exhibit—to see the power station, the electronics shop, the spare fusion equipment, etc.

The biggest crowds concentrated in the fusion section. There they could see not only operating fusion devices and models of future ones but also an array of equipment that represents the development of many new techniques and pieces of equipment, things that may well find important application in other areas.



POWER STATION supplying U. S. exhibit is viewed by NUCLEONICS Chief Editor Jerome D. Luntz, who led three-editor team covering Geneva

Fusion: Where Do We Stand?

GENEVA, *September 1–13*

One of the most important accomplishments of Geneva-'58 was that a true assessment of where the world stands in controlled thermonuclear power was made possible. It was constantly emphasized that a long road lies ahead to the accomplishment of three important stages: (1) a self-sustaining reaction, (2) a reaction that makes as much power as is required by the containment device and (3) a device that is a net power producer.

Meanwhile, much work needs to be done in plasma physics. The best approach is not obvious: all currently operating devices have their usefulness in plasma study. Possibly some devices are producing thermonuclear reactions on a miniscule scale now—and bigger things are around the corner. Artsimovich says: "When the number of neutrons arising in a single discharge pulse reaches a value in excess of 10^{12}, then all doubt as to the origin of this effect will vanish."

Fusion Time Schedule

The main question that was on everyone's mind was, "When will we have controlled fusion?" Evidence was available for the pessimists, the optimists, and middle-of-the-roaders. The optimists pointed to what may be thermonuclear neutrons from the American Scylla and the Russian ion magnetron and to the prediction that DCX may achieve burnout momentarily (*see pp. 67 and 69*). Meanwhile the pessimists pointed to some theoretical and experimental discouragements. Russian computations that take into account higher order terms that have not previously been evaluated indicate that radiation losses from truly hot plasmas may be much greater than has been anticipated (*2213*). Several American speakers, notably J. L. Tuck in his review of LASL work (*1860*), referred to a still unknown loss of energy (possibly the result of "cooperative phenomena") that is currently plaguing pinch experiments. (It is not the radiation loss previously referred to; plasmas have not yet reached the temperatures at which that becomes important.)

Tuck indicated that two explanations are available for the unknown loss: instabilities of the plasma as a whole and motions within the plasma. If the correct explanation turns out to be whole-plasma instabilities, something can probably be done about it. On the other hand, if the explanation is in plasma oscillations in which electrons and ions play different roles, the result may be a catastrophe without a solution. This would force the abandonment of the pinch effect as a potential power source.

The experts seemed to be taking a cautious, middle-of-the-road view. As the first predictor of ultimate fusion power and the most highly qualified unbiased observer, Homi Bhabha was the target of many questions on the time scale. Asked if he would alter his 1955 estimate of 20 years in the light of revelations at the '58 conference, he replied with a smile that he could now revise the estimate to 20 minus three. Edward Teller indicated that materials technology might be able to prepare for the challenges of fusion reactors in the 17 years Bhabha had predicted. P. R. Bell, answering questions about whether DCX would be a fusion reactor as soon as it achieves burnout, replied that while there were no obvious obstacles in the way now, they were almost sure to appear as the next steps come along.

Pure Science

The theme that talk about fusion power is premature ran throughout the fusion meetings. The feeling was that plasma physics must be studied and that there is much to be learned before the knowledge can be used.

An often-used simile was that fusion work is now like a baby learning to walk. Several steps occur without difficulty, but the next may be the one on which we fall flat on our face.

It was often suggested that the solution of the fusion problem may follow a big, unexpected breakthrough. Teller's reply to this sentiment was that the present state of knowledge is such that we are not prepared to recognize a breakthrough if it occurs. He predicted many years of studying plasma physics with the hope that we can "pick some flowers along the way." Asked to identify his "flowers," he said that his imagination is too limited to predict them but mentioned such things as electric switching as practical developments that can follow from plasma research.

Best Approach

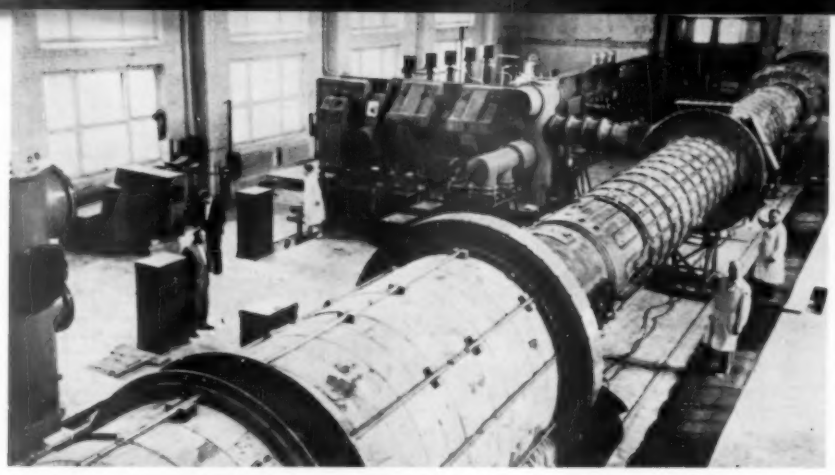
Everyone appeared convinced that his own approach to the fusion problem is the best. After one of the sessions on fusion devices a request for comments on which is the best approach brought a hearty laugh from all speakers.

Discouragement with the pinch did not appear to discourage the British, whose Zeta and Sceptre devices are based on pinch. They listened to all counter arguments. Asked why they stuck to the pinch and whether they thought a multiple approach might be better, they replied that a limited effort should be directed where it appears to have the greatest possibilities. Pinch, they repeated, appears to them to have the greatest possibilities. To a NUCLEONICS question as to whether political pressure at home might force diversification, they replied that they felt that they might be required to at least try some of the other approaches.

What Kind of Neutrons?

Everyone discussed his neutrons. Bell indicated that DCX makes them plentifully and it is plain that they are from wall impurities. J. L. Tuck and the Russians agreed that the question of origin is not important. Bell, Tuck, and the Russians indicated that when they get thermonuclear neutrons that are significant, they will get them in such numbers that there will be no doubt.

Meanwhile the Scylla and the ion magnetron people were pleased to say that there is no evidence yet that their neutrons are not thermonuclear ones.



OGRA, world's largest fusion device, was among Russian surprise revelations at Geneva

Fusion: The Russian Picture

GENEVA, *September 1–13*

Russia revealed a large and comprehensive fusion program at Geneva. Aside from its magnitude it had two total surprises for which (the few Russian papers received before Geneva gave no hint). One was their huge OGRA device "OGRA" stands for the initials of four of the men who worked on the project, which is surely one of the most elaborate and expensive fusion apparatuses of the world. The other was some detailed theory of radiation losses that show that more headaches are present in high-temperature plasmas than have been anticipated. Except for these things, British and American observers agreed that they had not learned anything new from what the Russians had to say.

The Russians were surprisingly candid. In private conversations they appeared to be holding nothing back. A paper by L. A. Artsimovich described their program in detail (2298).* He outlined the general theory of fusion as the Russians have developed it and described their experiments in three areas: fast pinch, stabilized pinch and plasma traps.

Multiple Approach

Like the United States and in contrast to the British, the Russians are pursuing thermonuclear problems along many paths. This was stated explicitly by Artsimovich. It was also apparent from their scientific exhibit and their private conversations with American fusioneers.

OGRA indicates that their biggest effort is on a magnetic trap with molecular-ion injection. It is roughly similar to the Oak Ridge DCX devices (*see p. 66*). Yet its size and the advanced stage of its development make it apparent that it was near completion before the Russians knew anything about DCX.

Private conversations with Russian delegates led DCX people to conclude that the Russians had quickly embarked on a program to duplicate DCX as soon as its details were published. The British came to a similar conclusion relative to their Zeta and Sceptre devices.

An operating linear-pinch device was part of the Russian scientific exhibit, their only actual fusion device on display. Meanwhile their theoretical papers showed that they are exploring twisted magnetic fields including stellarator-like figure-8's, radiofrequency containment like that considered at Argonne (*see page 68*), and radiofrequency traps at the ends of strong magnetic fields. They also have a rotating-plasma device that uses betatron acceleration.

OGRA Is Impressive

Looking at the great size of OGRA (see figure above) one is led to ask "Why?" One reason is that it must be big to make up for inadequacies in its injection and trapping methods. But it is apparent in both fission and fusion programs that the Russians are willing to risk big stakes by building big machines first. They are apparently prepared to skip the preliminary experiments that can be done with small devices. As one British expert phrased it, the eventual machine must be a big one, so perhaps it is good to start out with a big one.

Apparently OGRA is currently at the shakedown stage. This was the general conclusion of American observers from the evidence the Russians made available. Although, in general, the Russians were communicative, it was not easy to get an explicit statement on this point. This left unanswered the question as to how a large device can be designed, built and put in operation without any performance data becoming available.

Design. OGRA is a T-shaped device held high in the air on supporting legs. The cross bar is a 20-meter-long metal tube with a 1.4-meter internal diameter. Coils wound along the tube create a field of 5,000 gauss at the center, and mirror coils, separated by 12 meters, produce fields of 8,000 gauss.

The side tube, about 10 meters long, is for injection of D_2^+ ions into the trap. It contains an ion source that accelerates the ions to 200 kev and coils and electrodes for electric and magnetic focussing. Artsimovich stated that, "It is expected after adjustment of the apparatus it will be possible to raise the D_2^+ ion current entering the vacuum chamber to several hundreds of milliamperes." This should lead to D^+ concentrations of $10^{13}/cm^3$.

Ion insertion is always a problem. The Russians are attempting to solve it by inserting a metal snout into the

* References are to Geneva Conference papers; see p. 145 for list.

containing field. This distorts the field lines somewhat and permits the ions to enter the field with a magnetic energy low enough that they do not immediately escape. The disadvantage is that the snout is always present as a scattering target. Any ion in the system can be expected eventually to hit it and bounce out of the trap. Russians at the conference were eager for details of the extraordinary carbon arc in DCX. Some of the DCX people felt sure that such an arc would shortly become a part of OGRA for improved ion injection.

Artsimovich Paper

The Artsimovich paper at the plenary session on fusion gave a detailed description of the Russian fusion program. He discussed basic theoretical considerations and showed how the Russians were testing them with devices that fall into three classes: (a) fast pinch, (b) stabilized pinch and (c) traps.

Theoretical considerations. The paper makes it plain that the Russian theorists have discussed fusion in the same terms as the Americans and British. The basic dynamic equation that describes plasma behavior permits two basic simplifications. In one the assumption is that the plasma has a constant pressure at all points. This leads to pinch, in which electromagnetic forces balance plasma pressure. The other assumption is that there is negligible acceleration within the plasma. This leads to the concept of the magnetic trap.

Fast pinch. Early Russian tests of their theory were with fast-pinch apparatuses. Fast pinches were defined by the speaker as those in which the current rise time is short compared to motions in the plasma. The Russians have made detailed and cautious estimates of temperatures and densities in their plasmas and found values that agree well with their theory. Temperatures of 3–4-million °K were attained and maximum densities 30–40 times the original density of the gas.

They had no difficulty producing neutrons, but the speaker left to newsmen the worry about whether the neutrons were of the "noble race of descendants of thermonuclear reactions." He stated that when they could observe 10^{12} neutrons per pulse they would be convinced that the neutrons were thermonuclear. In the meantime he felt that it was not an important point.

The Russians are entirely pessimistic about the future value of fast pinches as power producers. They have calculated that in a power-producing pinch each pulse would have to be violent enough to destroy the apparatus containing it. Thus their present research in the subject is entirely for better understanding of plasma physics.

Other Pinch Experiments

Aside from straight-pinch devices the Russians have performed similar experiments with plasmoids and D-D and D-T explosions. Two types of plasmoids are developed. One is generated by the current in a loop. The resulting torus is guided by interaction with the wires leading to the loop. Observed velocities reach 5×10^7 cm/sec. In the other situation a toroidal plasmoid is generated in a magnetic field perpendicular to the plane of the torus. The torus then collapses into a blob that has particle density as high as 10^{16}/cm^3.

D-D and D-T explosions have been created by surrounding an ampoule containing the appropriate gas mixture with a chemical explosive and setting off the combination. The resulting explosions destroyed the measuring apparatus, but not before it had registered flights of both fast and moderated neutrons. The experiments have now been abandoned as having no practical value.

Stabilized pinch. Two kinds of stabilized pinches have been produced and examined. One is the paramagnetic pinch in which the magnetic field lines are held within the plasma and contract with it. The other, a nonparamagnetic pinch, occurs in a high magnetic field which stays relatively constant during the pulse.

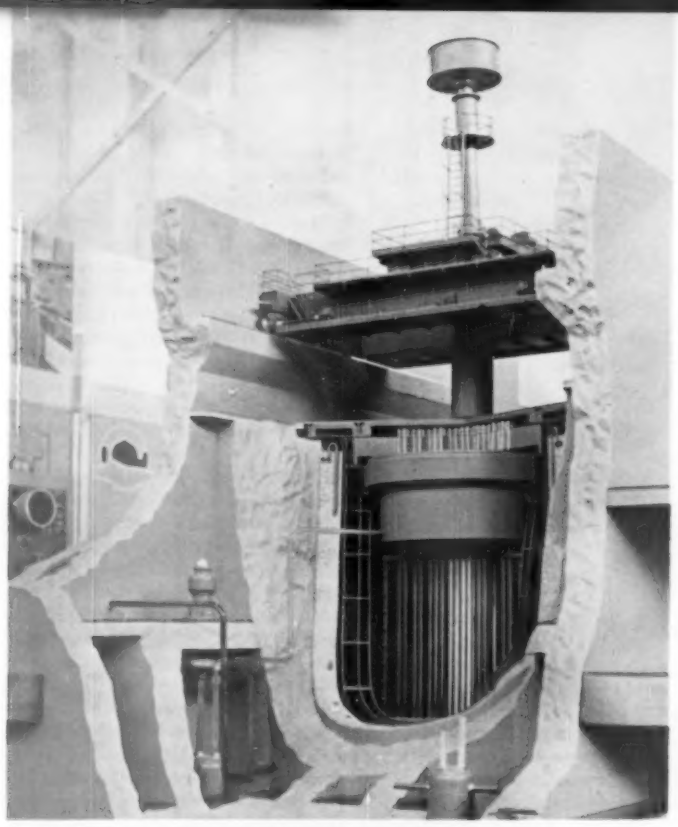
The paramagnetic pinch requires a conducting sheath around the discharge. Induced currents in the sheath reduce the magnetic field outside the plasma, and plasma currents cause the fields inside the plasma to build up as the plasma diameter decreases. Artsimovich quoted published British analyses of Zeta and expressed Russian agreement with them. The conditions he placed on axial and azimuthal fields are those that Americans and Britons have stated at the conference. He agreed also that the plasma must not contract too much lest it pull so far from the walls that currents in the surrounding sheath cannot perform their function.

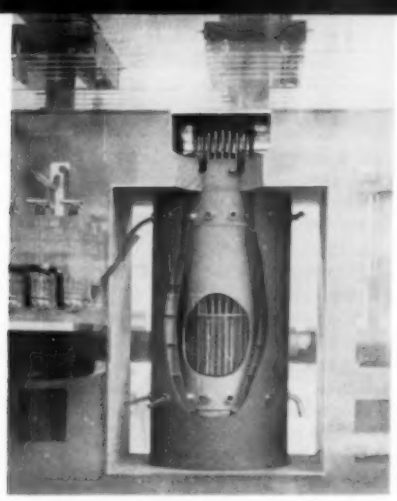
Nonparamagnetic pinches are found to have much promise. They may, however, require magnetic fields as great as 30,000–50,000 gauss. They are likely to suffer corkscrew instabilities, but these are not considered serious inasmuch as the axis of the plasma is not distorted. Artsimovich cautioned that the analysis that has been applied to these pinches is a simplified one. It is accomplished with magneto-hydrodynamic theory, which is inherently unable to deal with motions within the plasma.

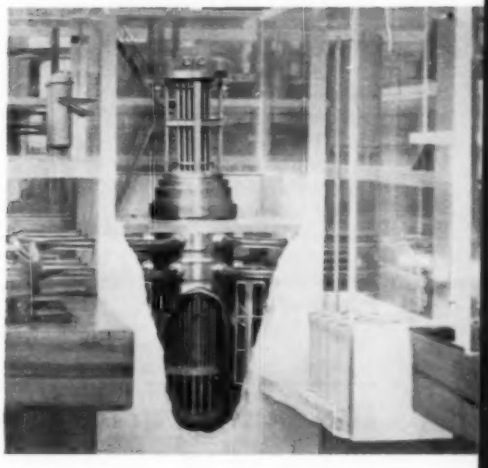
Russian experiments on nonparamagnetic pinches have progressed from linear discharges to toroids. The linear discharges suffered too large energy losses at the end electrodes. Artsimovich included a photograph and a cutaway drawing of a slow-pinch toroid much like the British Zeta and Sceptre. It has an inner diameter of 50 cm, and a mean diameter across the toroid plane of 1.25 meters. It functions as an air-core transformer with a primary coil consisting of 20 thick copper turns. The longitudinal field can be increased to 12,000 gauss, and a condenser bank stores 1.2-million joules. A peak discharge current of 400,000 amp has been attained.

Traps. Research in traps has led to the OGRA machine. Another device is an ion magnetron similar to the one recently revealed in the U. S. Here D^+ ions are accelerated from a central plasma to a surrounding wall by a 40,000-volt potential and caused to circulate by a longitudinal magnetic field. The vacuum chamber is 2 meters long, and the magnetic field varies from 8,500 gauss at the center to 12,000 gauss at the ends. Titanium pumps maintain a vacuum of 10^{-7} mm of Hg. Ion lifetimes of several milliseconds are predicted on the basis of experiments in which the accelerating voltage is switched off and plasma disappearance is observed.

Two other kinds of trapping are being investigated. One has a corrugated field produced by a series of conductors outside the plasma. The resulting inhomogeneity induces axial circulation of plasma particles and reduces escape. The other is r-f trapping. Containment by purely r-f fields has been investigated, and in experiments r-f fields have been used in conjunction with constant magnetic fields. The result is a trap in which an r-f plug replaces the usual magnetic mirror.







RUSSIAN REACTOR FAMILY exhibited at Geneva included models of proposed 50-Mw(e) sodium-graphite plant (above), 150-Mw(e) D_2O-CO_2 design (upper right) and 210-Mw(e) pressurized-water type (lower right). Latter will use 1.5% enriched UO_2 and is expected to begin operation in 1961. Two pairs of PWR design will be built to power central stations near Voronezh and Leningrad. Russian program also includes work on boiling water, homogeneous, fast breeder and graphite-H_2O concepts

Geneva Reviews Reactor Progress

GENEVA, *September 1–13*

A large part of Geneva dealt with power-reactor design and experience. Qualified observers disagreed sharply as to the over-all quality of the presentations. Some felt that a good share of the papers —Western as well as Russian—were uninformative in the sense of remaining silent on important points; there was also heard some dissatisfaction with the overwhelming number of papers presented, as well as with their selection and the need of meeting the demands in this regard of "politics" (that is, the requirement of geographic distribution).

Others, on the other hand, thought that technically this conference was better than its 1955 predecessor; that the general feeling of anticlimax that pervaded the corridors should really be ascribed to the fact that fission reactors are no longer automatically thrilling as they were three years ago, and that if the excitement of 1955 was lacking, this was only a measure of the distance nuclear power has traveled on the road to becoming a conventional engineering technique.

One thing seemed certain, however: although there were numerous reports on novel reactor engineering details, novel solutions to familiar problems, and some ingenious avenues for future development, there was not a single new reactor concept unveiled. Coming closest were the British Advanced Gas-Cooled Reactor (AGR) and the High Temperature Gas-Cooled Reactor (HTGR) at Winfrith Heath; the German pebble-bed reactor being built by Brown Boveri and Krupp; and a Russian zero-energy uranium hexafluoride gas fueled reactor. (For description of similar concept by an American see NU, Aug. '58, 128).

Russian Reactors

First sensation of the conference's first eight days was a dramatic revelation by the head of the Soviet delegation, V. S. Emelyanov, that on the Sunday midway through the conference his country had brought to full power a hitherto unrevealed 100-Mw(e), dual-purpose power and plutonium reactor in Siberia. Graphite-moderated, light-water-cooled and natural-uranium-fueled, it is a scale-up of the now-famous Russian 5-Mw(e) "first atomic power station" disclosed with similar fanfare at the 1955 Geneva conference.

A 17-minute color film showed delegates a tremendously spacious plant with reactor hall much like that of the smaller prototype—the reactor top is in the center of the floor, spare fuel elements hanging on the walls. Fuel elements are pure natural uranium metal rods, aluminum clad, six meters long; they

are loaded by cranes from top (channels being closed by hand) and are unloaded automatically from bottom into a pool of water. A massive shielded trouble-shooting machine containing two men moves in horizontal plane between reactor bottom and the pool, and keys under any channel in case the automatic unloader becomes jammed.

Primary water enters at 180° C and leaves at 220° C; secondary system delivers 185°, 90 lb/sq in. superheated steam to three turbo-generators. Fuel loading is about 200 metric tons natural U. Design cost operation is eight to ten kopecks/kwh, said Emelyanov.

Reactor is first of six units at what is to be a 600 Mw(e) nuclear station serving a new Siberian city built simultaneously with the reactor and not yet named. Emelyanov turned aside persistent questioning regarding its location saying it is in a "completely new undeveloped region having no geographic identity." He said the power will be used for industrialization of Siberia. He told a questioner the reactor has reached full power.

The Russian reactor effort, as it unfolded day by day, added up to a very impressive picture—and top U. S. reactor designers said frankly that they were impressed.

Russian work on pressurized water reactors appears to parallel U. S. work very closely. The work is extensive and quite good but not more advanced than U. S. work. Little was presented that was new to American PWR developers except for small details; however the little bit of information furnished on aluminum cladding seems better than Americans have been able to do. A quite detailed description was given of the two 210-Mw(e) pressurized water reactors being built for a 420-Mw(e) nuclear power station at Voronezh, and the reasons behind the choices of major plant and reactor parameters were discussed in some detail. Variants of the plant using direct-cycle and dual-cycle boiling were also considered. This plant, already described at Belgrade last year (NU, July '57, 20) appears to be a conventional pressurized water type with no novel or interesting features; the level of reactor physics work reported was not found to be sophisticated in any sense, to the surprise of the U. S. experts.

Besides the Siberian dual-purpose reactor the Russians described another 100-Mw(e) graphite-moderated and H_2O-cooled unit which is optimized for power. The design is based on experience in operating the first, 5-Mw(e) station and is of the same type. As a result of experiments in the 5-Mw unit with controlled boiling in certain channels, the 100-Mw(e) scale-up will not only use a direct steam cycle, but will superheat the steam by recycling it through special fuel elements in the reactor—seemingly a substantial advance in reactor technology. About ⅔ of the fuel elements have water entering them and are in a bulk boiling condition. The steam thus produced goes to a heat exchanger and is then condensed; the steam formed in the heat exchanger is dried and fed through the remaining ⅓ of the core, producing a substantial amount of superheat. Efficiency, based on gross electrical output, is placed at

Russian Reactor Program

Reactor	Fuel	Moderator	Coolant	Power	Location	Date of criticality	Steam at turbine
First station	5% enriched U	graphite	H_2O	5 Mw(e)	Obninsk	27 June 1954	
First large station	nat. U rods stainless clad	graphite	H_2O	600 Mw(e) (6 reactors)	somewhere in Siberia	7 Sept. 1958 (full power reached)	90 PSI, 185° C
Uranium-graphite	1.3% enriched, cladding undetermined	graphite	H_2O	400 Mw(e) (4 reactors)	Beloyarsk in Urals	?	1,320 psi, 480–500° C
PWR	1.5% enriched UO_2 rods Zr-clad	H_2O	H_2O at 1,470 psia	420 Mw(e) (2 reactors)	Voronezh district	1961	430 psia, sat
PWR	"	H_2O	H_2O at 1,470 psia	420 Mw(e) (2 reactors)	Leningrad district	?	
BWR	" (144 elements)	H_2O	H_2O at 104° C inlet	50 Mw(e)	Ulyanov district	?	480 psi, 230° C
Homogeneous	UO_2 suspension in D_2O	D_2O	fuel suspension	35 Mw(th)	on Volga	?	
Sodium-graphite	different types will be tested	graphite	Na at 118 psia	50 Mw(e)	on Volga	?	1,320 psi, 500° C
BN-50 (fast breeder)	Pu-U alloy and Pu alloys with Ba, Fe, etc., will be tested	none	Na (at 500° C max.)	50 Mw(e)	on Volga	?	
BN-250 (fast breeder)	Pu	none	Na (at 540° C avg.)	250 Mw(e)	not yet decided	?	
Intermediate-neutron	?	?	?	50 Mw(th)	?	?	
Mobile PWR	?	H_2O	H_2O at 1,764 psi	2 Mw(e)	Obninsk	?	290 psia, 280° C
Lenin (icebreaker)	5% enriched, sintered UO_2 rods carried in Zr alloy	H_2O	H_2O at 2,940 psi	270 Mw(th) (3 reactors)	shipboard	entry into service spring 1959	430 psi, 310° C
Gaseous fuel	UF_6 gas	Be metal	uncooled	0	?	1957	

35% or greater. The reactor was tested on a full-scale critical experiment.

The fuel element is basically like that in the first prototype power station: tubular, of stainless steel containing stacked graphite disks pierced by five vertical holes (a central and four narrower peripheral holes); through each hole passed a thin-walled stainless steel tube. Coolant water flows down the central hole to a small header at the bottom of the fuel element and up the four peripheral tubes, around each of which, in an annulus within the graphite, is packed 5% enriched uranium alloy. Coolant is piped individually—and can be controlled individually—to each of the elements, of which there are 128 in the 5-Mw(e) reactor, 998 in the 100-Mw(e) reactor (730 boiling and 268 super-heating elements).

Burnup is expected to reach 2,000–3,000 Mwd/t with fuel shifting. Calculated lifetime of the elements in the small reactor was 100 days, but four elements, left in longer to test lifetime, were still serviceable after 350 days (23 kg/t specific burnup).

Fast reactors. Russian work on fast reactors struck Americans as particularly impressive. A series of five were described (see table, p. 28): BR-1, for physics constants work, was started up in early 1955; it had no forced cooling and operated at 10 watts. BR-2, 100 kw(th), was built to establish the coefficient of breeding; like the smaller unit, it has 10 mm dia, 130 mm high plutonium rods in the core and a depleted uranium blanket. It is mercury cooled, and was started up in early 1956. (Mercury was chosen as coolant because of "complete absence of working experience with sodium-potassium eutectic" at that time; it is not a prospective heat-transfer medium for power reactors, it is specifically stated.) BR-5, a 5-Mw(th) unit with sodium cooling and fluxes to 10^{15}, will be next and will do research on fuels and materials to bridge the gap from BR-2 with its mercury cooling to a commercial-size power producer. Now in the design stage are BN-50, a 50-Mw(e) experimental power unit, and BN-250, a 250-Mw(e) reactor whose construction will start only after BN-50 is in operation.

Knowledge of plutonium and plutonium oxide metallurgy, corrosion and fabrication technology impressed U. S. observers. Breeding ratios—

NIOBIUM-zirconium-clad UO_2 rods in hex array form fuel assembly for USSR 210-Mw(e) PWR. Maximum design heat flow for 91 rod assembly is 1,200 cal/m² hr

2.5 ± 0.2 for BR-1, 2.1 ± 0.3 for BR-2, 1.9 for BN-50 and 1.8–2.0 for BN-250 (none was given for BR-5)—seem optimistic, but may be evidence of Russian design competence.

The Russians described their comprehensive program of work on zirconium and its alloys as regards corrosion resistance, mechanical properties, gas pickup, etc. This appeared to U. S. observers as a redevelopment of existing U. S. technology in many respects, but a contribution in the sense of confirming much known material. The most significant difference seems to be introduction of niobium into their complex, zircaloy-like alloys, for which good corrosion-resistant is claimed in both water and steam at high temperatures.

Icebreaker. Interest ran high in the icebreaker *Lenin* (see page 92) and it was speculated that the Russians have nuclear submarine capability; Americans in position to know said the *Lenin* reactor's resemblance to *Nautilus* plant was striking. One of them remarked that it was clever of Russians to put their prototype in an icebreaker where it would do some good instead of being stuck away in an Idaho desert.

Two reactors supply all the steam for propulsion plants at 42,000 hp output; third is spare but is identical to the others. The three are arranged side by side—not fore and aft—and with their vertical steam generators are located in one stainless steel shielded container. Each reactor with two complete primary loops forms an autonomous steam raising plant. Reactors are three pass, like *Savannah*, with coolant entering at bottom. Flow rate is 1,000 cu meters/hr.

Core is about 1 meter diam by 1.6 meters high; loading per reactor is 85 kg U^{235} at enrichment of 5%. Burnable poison, "natural mixture of boron isotopes," is introduced into central channels only. The maintenance of power level and load transients is performed by automatic regulators consisting of three rods; there are two such regulators, one being reserve. No rods for manual control are provided. Safety rods—their number unspecified—drop in 0.6 sec. Control material was not given.

The weight of atomic plant excluding shielding is 1,054 tons; shielding weight 1,963 tons; weight propulsion plant 2,750 tons; specific weight 68.5 kg/hp.

Endurance at top speed is two to three years as each core is designed for full capacity operation for about one year. The head must be pulled for fuel element changing but it is described as an easy job.

The reactors are to be loaded after the conference, final tests conducted in October, and "in spring we hope it will start piloting ships" across the Arctic Sea.

A bold concept was put forward by Emelyanov in his general talk on the Soviet program. Speaking of deuterium and tritium fusion he said: "Modern engineering possibilities show that a reactor with energy emission density of millions of kilowatts per cubic meter of plasma is a real thing. This value is higher than that in any other power unit including nuclear reactors and jet engines. Thus in principle it is possible to design deuterium-tritium reactors sufficiently compact and useful to be installed, for instance, on ships." At a press briefing Emelyanov declined to place a timetable on this development: "Such a reactor has not been built but solution of the problem appears likely."

Experience. Three subjects the Russians steadfastly refused to answer questions on were target completion dates for reactors, estimated costs, and Soviet uranium reserves. One break on costs came when S. A.

Design Data for Russian Fast Breeders

Parameter	*BN-50*	*BN-250*
Electric power [kw(e)]	50,000	250,000
Core diameter (cm)	65	107
Core height (cm)	65	107
Shield thickness (cm)	60	60
Average thermal flux in core (kcal/m²/hr)	1.8×10^{6}	2.5×10^{6}
Average specific power [kw(th) core]	800	1,000
Heat-transfer agent	sodium	sodium
Total Na flow rate (m³/hr)	3,500	—
Average Na velocity in core (m/sec)	6	9
Average Na output temperature (° C)	480	540
Steam temperature (° C)	415	—
Steam pressure (kg/cm²)	29	—
Maximum neutron flux (n/cm²/sec)	9×10^{15}	1.3×10^{16}
Breeding ratio	1.9	1.8–2.0
Core volume fractions		
Sodium	0.4	0.44
Fuel elements	0.47	0.43
Structural materials	0.13	0.13

Skvortsov said cost per installed kilowatt for big pressurized water plants is estimated to run 20–30% higher than an equivalent conventional plant. The operating cost of this plant was estimated at seven kopecks/kw [at the official four-rubles-per-dollar rate this is 17½ mills] whereas conventional power costs vary from five to twelve kopecks.

In describing four years' experience with their first atomic station the Russians mentioned four incidents of fuel element damage: two cases of leaks in tubes feeding water to an element, one case of rupture of the outer shell on an experimental element being tested, and one case of operator error in permitting flow water to one element to drop below tolerance with consequent overheating and rupture of both inner and outer tubes. In the last case fission products penetrated the graphite brickwork lattice but the channel was washing and there was no damage to graphite. Contamination was confined to reactor core, did not even spread to water in loop, it was said. In all cases the warning systems were reported to have operated normally.

Russians also reported in detail on their research reactors, describing seven types. Reconstruction of three older ones to increase power and flux levels as well as construction of three new ones, was recounted. Under construction is a large intermediate neutron flux trap reactor to be completed next year. The flux is $2.5–3 \times 10^{15}$ and will be the highest of any Russian research reactor.

Western Reactors

By contrast with the Russian revelations, western reactor papers and discussions furnished a treasuretrove of technical details but no massive disclosures. Only two new major reactor projects:

- American Machine & Foundry has a power reactor project again. Cuba has scheduled a 22 Mw(e) boiling water power plant at Santa Lucia harbor in Pinar del Rio province, west of Havana. The Cuban National Bank for Social Economic Development completed negotiations with the Anglo-American partnership Mitchell Engineering of London and AMF, with Zinn's General Nuclear Engineering as designer. The plant is to operate in 1961 to serve three new industries in the area.
- France's Atomic Energy Commission announced construction of a new developmental power reactor—EL-4—to test the heavy-water-moderated, gas-cooled concept.

Other highlights:

- Official secrecy still shrouds the choice at this writing of the SENN group in Italy—a choice due to be announced "mid September"—but the grapevine here seems widely convinced that it will be a General Electric boiling water station. In a searching analysis of power costs for Southern Italy Corbin Allardice of the World Bank concluded that an atomic plant would be very close to, if not exactly, competitive with oil-fired. He considers the enriched water and natural-U, gas-cooled units proposed as essentially on equal footing, with the former the best at 11.1 mills and the latter 12 mills.

Other encouraging cost news came from Samuel Untermyer of General Electric regarding the Vallecitos reactor. He estimates power cost using fully enriched fuel at about 35 mills at its present rating of 20 Mw(th) whereas calculations show that the present 3-ft high core would operate stably at 80 Mw and still higher outputs are possible with larger cores (reactor can take cores up to 4½ ft). Despite experimental changes the plant has generated 7.3-million kilowatt hours through September 1; it is now testing a core of ¼-in. stainless tubes filled partially with enriched UO_2 powder.

- United States heavy water production was revealed to be 1,000 tons yearly.
- Britain's Sir Leonard Owen declared that there has been no firm decision to convert later civil power stations to make military plutonium, as had been widely understood. He explained that the U. K. Atomic Energy Authority merely decided to install additional fuel charging and discharging capability to make that possible if it should subsequently appear desirable.
- In addition to Britain's AGR to be built at Windscale and HTGCR to be developed at Winfrith Heath (*see page 78*) the U. K. also has under development a heavy-water-moderated, gas-cooled reactor (comparable to France's aforementioned EL-4) as a hedge against AGR's being disappointing. Its attractiveness lies in the promise of low enrichment, no containment, possible operation with natural U, and plutonium recycle. Main emphasis in the study now is on engineering aspects, including header design, moderator cooling and insulation from hot coolant gas, and heat transfer. If a satisfactory design can be evolved the first commercial station might be commissioned late in the 1960's.

OTHER NEWS

While the eyes of the nuclear world were on Geneva last month, news continued to be made elsewhere. These developments are covered in a news section beginning on page 152.

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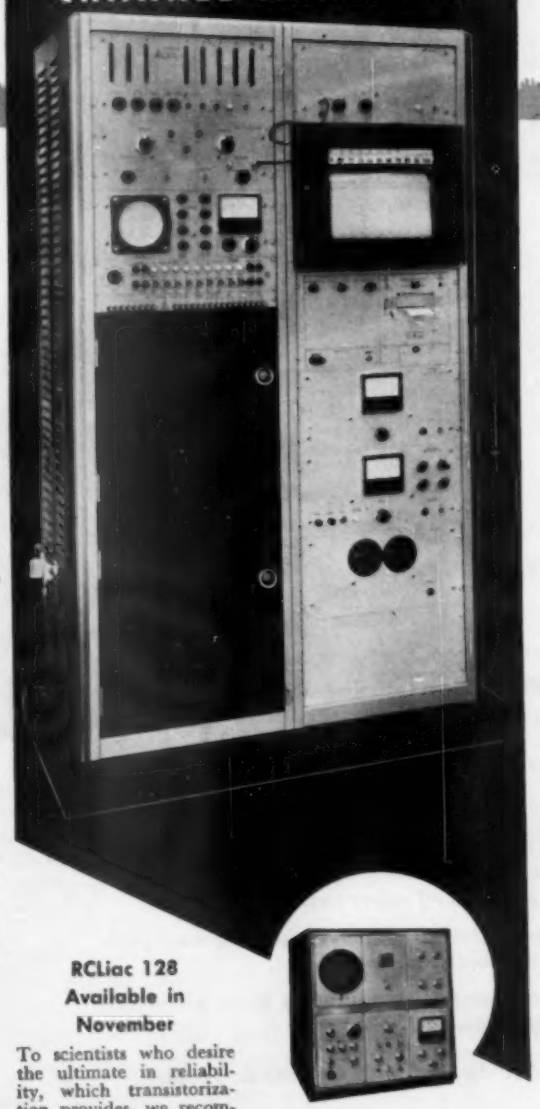

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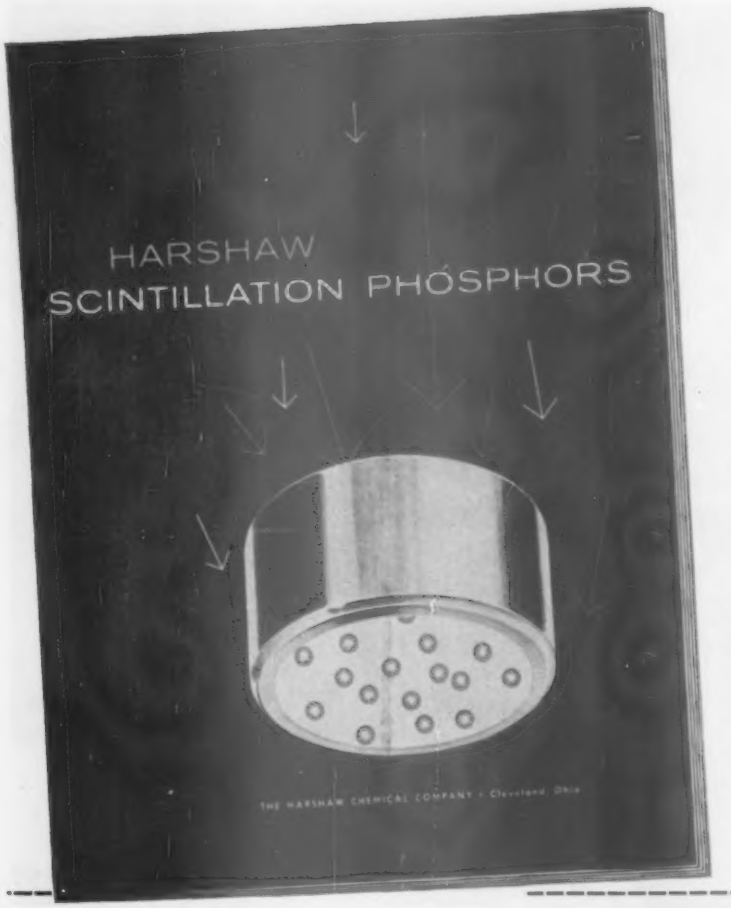

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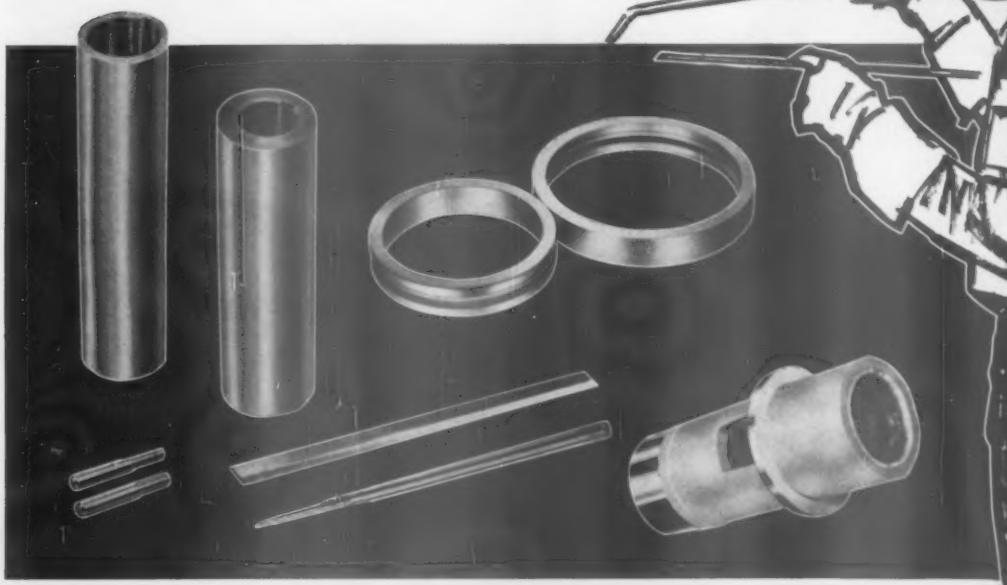

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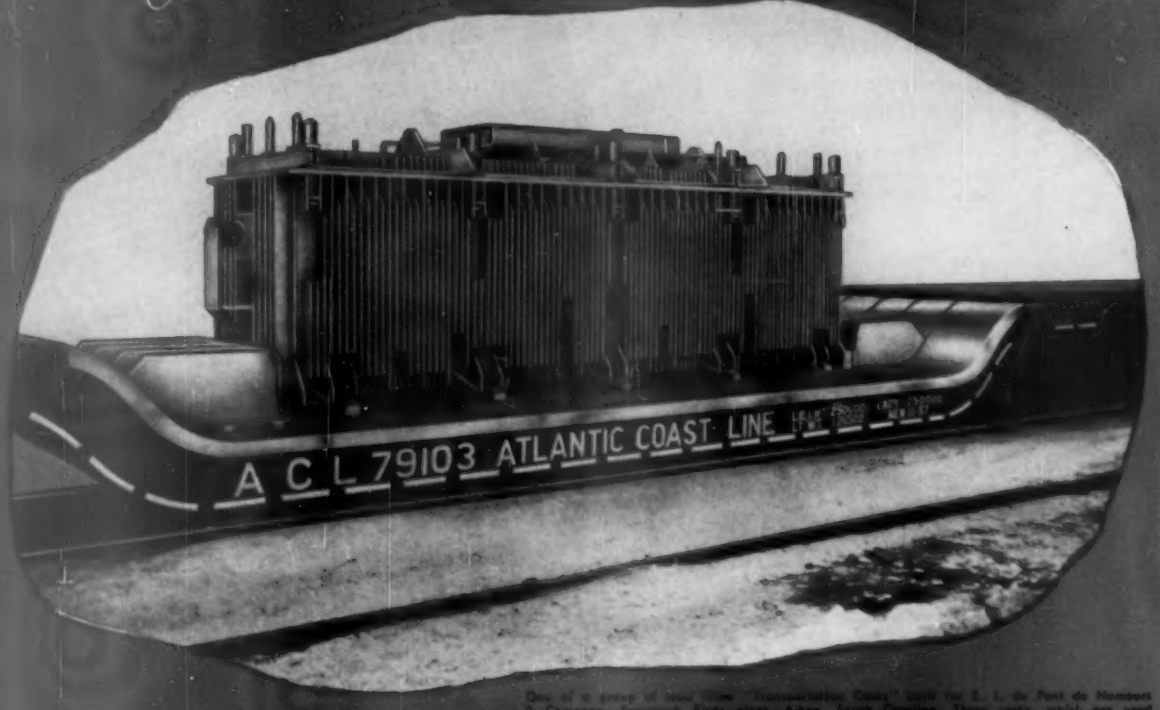

One of a group of lead lined "Transportation Casks" built for E. I. du Pont de Nemours & Company, Savannah River plant, Aiken, South Carolina. These casks, which are used for the shipment of radioactive materials, are fabricated from stainless clad steel with a 4½" thickness of lead homogeneously bonded to the interior carbon steel surfaces for heat transfer and radiation shielding. The hundreds of stainless steel "fins" on the exterior surfaces permit the dissipation of heat.

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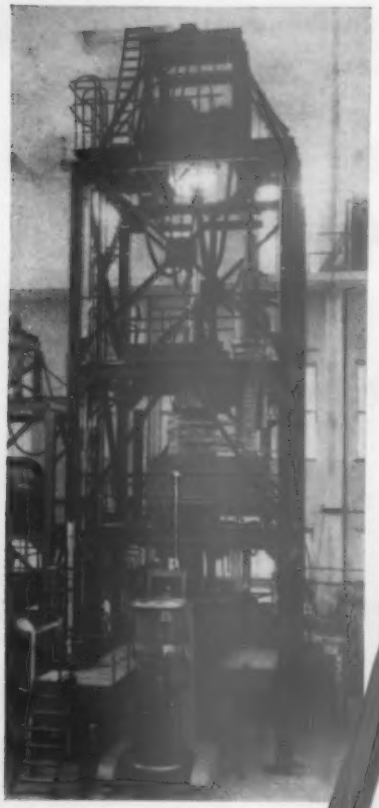

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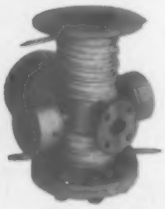

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Engineering Development Center Handling Procurement, Construction and Off-site Operation of Test Loop

The Lummus Company Engineering Development Center, under a subcontract from Knolls Atomic Laboratory, has completed the procurement, construction, and off-site operation of a test loop to be used for irradiation testing by the Knolls Atomic Power Laboratory.

Lummus simulated the confinements of the area assigned to the facility. All out-of-pile equipment was erected, although the in-pile test section was simulated.

The facility consists essentially of a pressurized water heat exchange system and demineralizing equipment. Reactor fuel samples are inserted in the in-pile test section for study under controlled conditions of temperature, pressure and radiation.

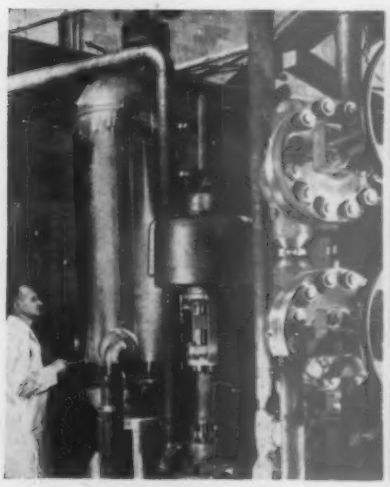
Extreme care has been exercised during fabrication and erection because of the severe operating conditions encountered and the cleanliness required in all equipment subject to radioactivity.

The off-site testing consists of checking the functions of all process and control equipment, instrumentation systems, safety interlocks and alarms.

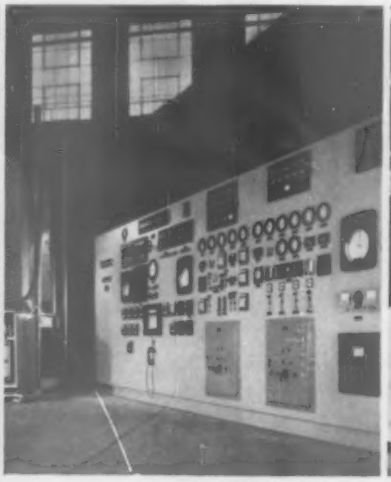
After completion of the test operation in early July, 1958, the equipment was dismantled and shipped to Idaho Falls. The dismantling was handled in a manner which facilitated re-assembly at the operating site, and Lummus provided the services of an engineer to coordinate this phase of the program.

The facilities of The Lummus Company Engineering Development Center in Newark, N. J. are uniquely suited to unusual projects of this nature. The necessary floor space, headroom, materials handling equipment, transportation facilities, and utilities are all available at the Center. Lummus' technical, maintenance, and operating personnel are all skilled in design, fabrication and operation of pilot plants and small scale process equipment. For more information write for our illustrated brochure No. 106, on the "Engineering Development Center."

THE LUMMUS COMPANY, 385 Madison Avenue, New York 17, N. Y., Houston, Chicago, Washington, D. C., Montreal, London, Paris, The Hague, Caracas, Maracaibo. *Engineering Development Center,* Newark, N. J.



Pressurizer, main flow control valve, loop block valves.



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Loop pump volutes prepared with special cover plates for hydrostatic testing.

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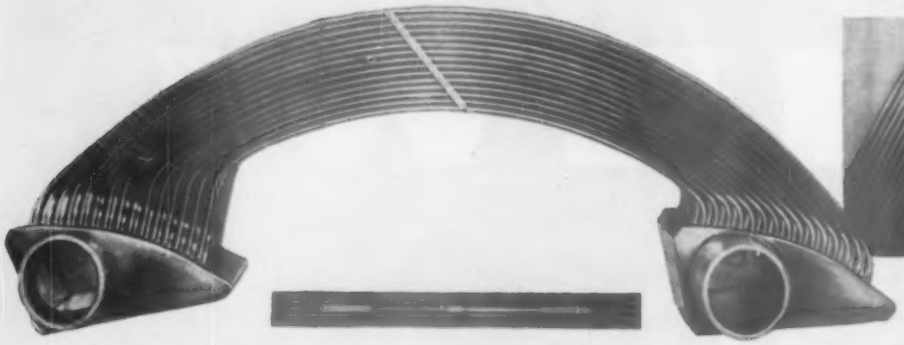

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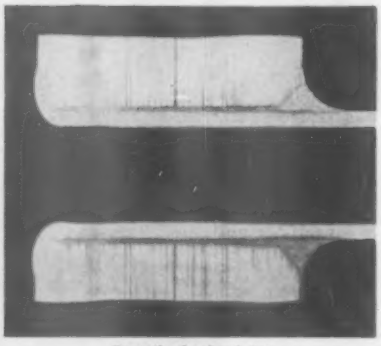

Detail of tube joint

Photograph of G-R's brazing furnace facilities

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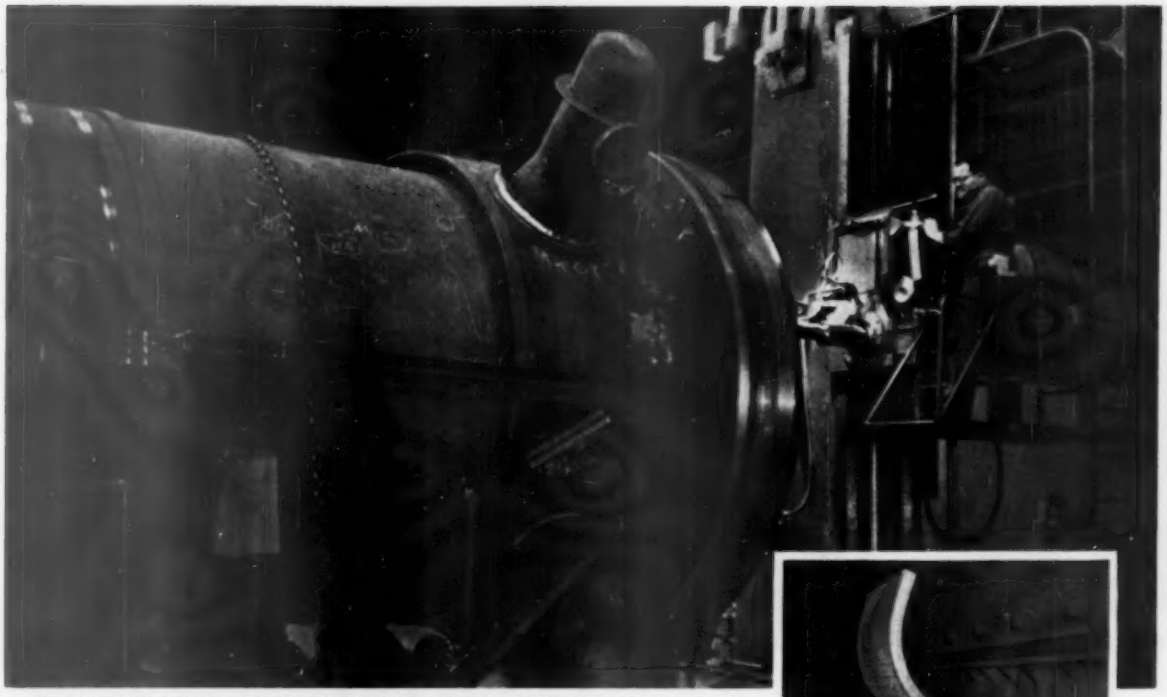

The 100-ton Foster Wheeler reactor vessel, with its two "halves" welded together, almost dwarfs the huge horizontal boring mill which is being used to bore stud holes in the flange.

Bending the thick steel plate on the 8,000-ton beam press.

Automatic weld cladding machine bonds stainless steel to the flange surface.

Flange and first section of vessel, ready for twin wire submerged arc welding of circular seams.

Foster Wheeler fabricates a

100-TON REACTOR VESSEL

for land prototype of Navy's first *NUCLEAR-POWERED SURFACE SHIP*

THIS huge pressure vessel—one of the largest nuclear components ever built—was designed and fabricated by Foster Wheeler for the prototype of the nuclear propulsion plant for the Navy's first nuclear-powered surface ship. This prototype plant is being constructed at the U. S. Atomic Energy Commission's Naval Reactors Facility, Idaho Falls, Idaho. The work was done under subcontract to the Bettis Atomic Power Division of Westinghouse Electric Corporation which is responsible for the design and development of the nuclear propulsion plant for the prototype under the direction of and in technical cooperation with the Naval Reactors Branch, U. S. Atomic Energy Commission. The pressure vessel, one of two to be used in the prototype, will house a reactor core or "furnace" where the nuclear reaction or fission process takes place.

The shell was hot formed on the 8,000-ton beam press from thick manganese-molybdenum alloy steel to which stainless clad was bonded. The flange, machined from one of the largest one-piece forgings ever made to this alloy specification, was clad with stainless steel deposited by a new automatic process developed by Foster Wheeler. The vessel was fabricated in two shell sections, one including the flange and the other, the lower head. The completed sections were welded together just before the final machining operations.

All seams were welded by a special submerged arc process developed by Foster Wheeler, and all critical welds were inspected by radiography. This was done with a 2,000,000 volt X-ray generator which is capable of penetrating 12 inches of solid steel.

Foster Wheeler also designed and built the stainless steel thermal shields which stand vertically inside the vessel; the vessel supports, and the insulation retainer assembly.

This giant reactor vessel is an excellent example of Foster Wheeler's ability to fabricate large, high-pressure cylinders and weldments to the most exacting specifications required by industry.

Foster Wheeler Corporation
666 Fifth Avenue, New York 19, N.Y.

FOSTER WHEELER

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Photo courtesy Atomics International

THIS SODIUM-COOLED REACTOR IS CHARTING THE WAY TO ECONOMIC NUCLEAR POWER

One of the most promising routes to economic nuclear power — via a sodium-cooled, graphite-moderated reactor — is being explored in the Sodium Reactor Experiment (SRE) at Santa Susana, California, by Atomics International for the A.E.C.

For more than a year, SRE has supplied heat to generate electricity for a commercial grid on an experimental basis. Designed for 20,000 thermal kilowatts, the reactor produced 21,000 kilowatts of heat in its most recently reported power run. However, SRE's principal output is design, construction and operating data for building full-scale nuclear power plants.

Key to the economic significance of the sodium-graphite reactor is its high thermal efficiency — approaching that of conventional steam power plants. Sodium's high boiling point makes this possible by permitting high-temperature operation at low pressure. At full power, liquid sodium now leaves the SRE reactor at 945° F and 3 psig to generate steam at 825° F and 600 psi through heat exchange. Moreover, a new reactor core will be installed to raise sodium temperatures substantially, and increase efficiency.

Liquid sodium has these other basic advantages as a reactor coolant: low neutron absorption, high heat-transfer efficiency, extreme thermal stability, ease of pumping, freedom from corrosion in the absence of impurities.

For further information on sodium, write for our brochure, "Handling Metallic Sodium on a Plant Scale." And call on us for technical service if you have specific problems.

FIVE OTHER
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- Experimental Breeder Reactor-1 (EBR-1) NRTS, Arco, Idaho — fast breeder, sodium-potassium cooled — 15,000 KW (operational)
- PRDC Fast Power Reactor, Monroe, Mich.— fast breeder — 100,000 KW (under construction)
- Consumers Public Power District, Hallam, Neb.— graphite moderated — 75,000 KW (design stage)
- Experimental Breeder Reactor-2 (EBR-2) NRTS, Arco, Idaho — fast breeder—20,000 KW (design stage)
- Chugach Electric Association, Anchorage, Alaska—heavy water moderated — 10,000 KW (development stage)

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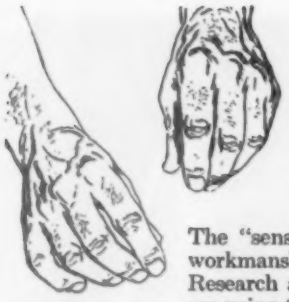

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The "sense of feel", the versatility, the uncompromising workmanship . . . all are found in every CRL Manipulator. Research and production organizations all over the world are using these units for handling radioactive materials, explosives, and hazardous chemical and biological materials. Natural motions of the operator's hands are copied exactly at the slave end. The "sense of feel" helps the new operator to become perfectly competent with practically no training period. Handles are adjustable to comfortably fit a wide range of hand sizes.

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2. *Flexible booting* Protects slave end from dust and contamination. Restricted gas-flow seal possible. Won't interfere with manipulator's natural movements.

3. *Special-purpose tongs* For repetitive or highly specialized operations. Wide selection of types to meet all requirements.

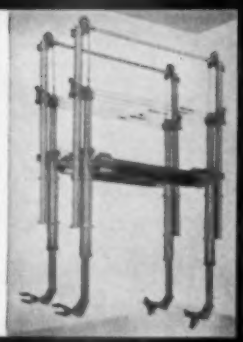
accessories
for master-slave manipulators

4. *Load hook* Available for CRL Model 8; may be remotely attached or removed. Permits a vertical lift of 60 lbs.

5. *Motion locks* Permit objects to be rigidly positioned in space. *Locking grip* standard on all models. Locks for all other motions available as accessories.

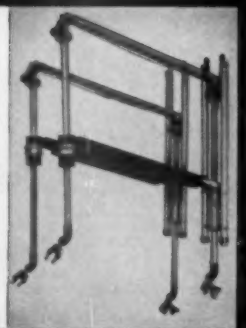
CRL Model 4
for over-wall installation

A general purpose manipulator for installation over a barrier wall or through a ceiling opening of a protective enclosure.



CRL Model 7
for restricted space installation

A space conserving, light weight manipulator for use in hoods or over thin walls where minimum clearances and easy portability are important factors.



CRL Model 8
for thru-wall installation

A general purpose manipulator for installations requiring intermediate to high level shielding, the Model 8 features

- *Accurate "sense of touch"* Low friction forces . . . low backlash . . . low inertia
- *Greater versatility* A variety of tools, tongs and locking motions available
- *Easier sight control* Operator need not step back to bring materials to barrier window
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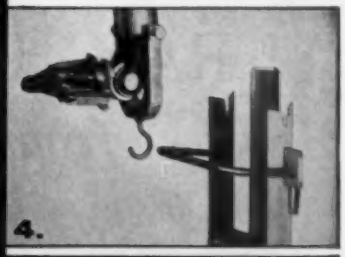

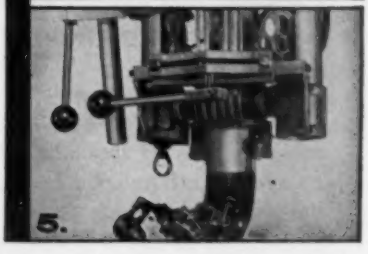

STRAIGHT TALK TO ENGINEERS

from Donald W. Douglas, Jr.

President, Douglas Aircraft Company

I've been asked whether non-aeronautical engineers have good prospects for advancement in the aviation industry.

The answer is *yes, definitely!* At Douglas many of our top supervisory people have moved up from other engineering specialties. The complexity of modern aircraft and missiles requires the greatest variety of engineering skills known to industry.

For example, we now have pressing needs for mechanical, structural, electrical and electronics engineers in addition to aerodynamicists, physicists and mathematicians. Whatever your background in the engineering profession may be, there are prime opportunities in the stimulating aircraft and missiles field.

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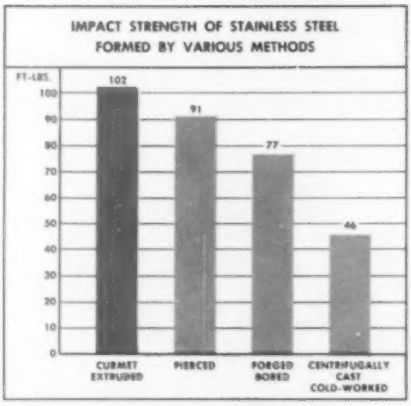
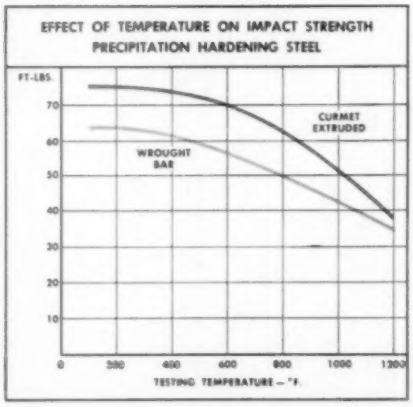
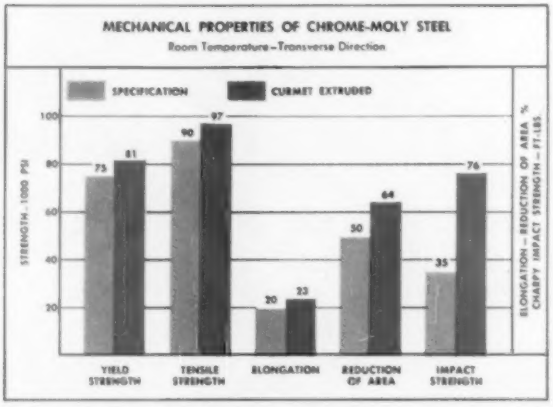


Impact strengths (Charpy) at 300° F. below zero. Curmet extruded Grade 304 stainless steel shows strength from 13% to 120% higher than the same alloy processed by other methods. Improvement in impact strength is consistent up to 1200° F.



Impact resistance of strength-at-high-temperature steels (A-286 in this case) is substantially greater when Curmet processed. The Curmet extrusion tests consistently 10 ft.-lbs. better than the usual wrought product.



"Above specification" characterizes the physical properties of all Curmet products. But impact strength is particularly noteworthy—more than 100% over specification—as shown by these test results on heat treated Grade P-22 steel.

Curmet Processing Adds to Alloy Performance

This is one of many outstanding achievements in metals from the Curmet plants and laboratories. As the frontiers of metallurgy advance under the pressure of more critical operating conditions, Curmet processing utilizes skills and knowledge, based creatively on wide experience with high performance metals, to give added integrity to new alloys.

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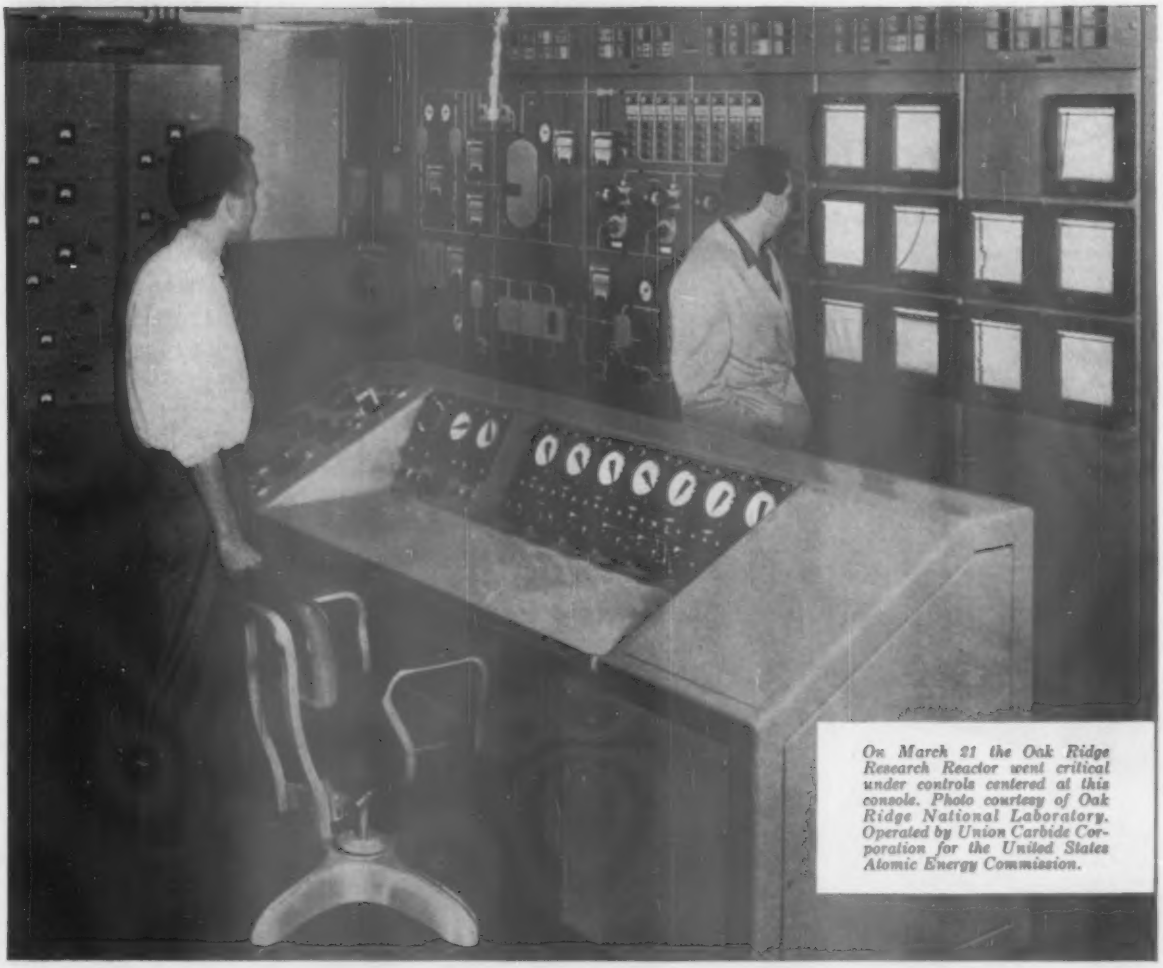

On March 21 the Oak Ridge Research Reactor went critical under controls centered at this console. Photo courtesy of Oak Ridge National Laboratory. Operated by Union Carbide Corporation for the United States Atomic Energy Commission.

Speedomax® H recorders concentrate flux and power level data

UP FRONT

for operators of ORR

For the new high-flux research reactor at Oak Ridge, an instrumentation system was specified that would present accurate, easy-to-evaluate records during all phases of operation. Clear, precise traces of changes in eleven important variables, including log count rate, log N, log N period, and safety systems, had to be directly before the operator.

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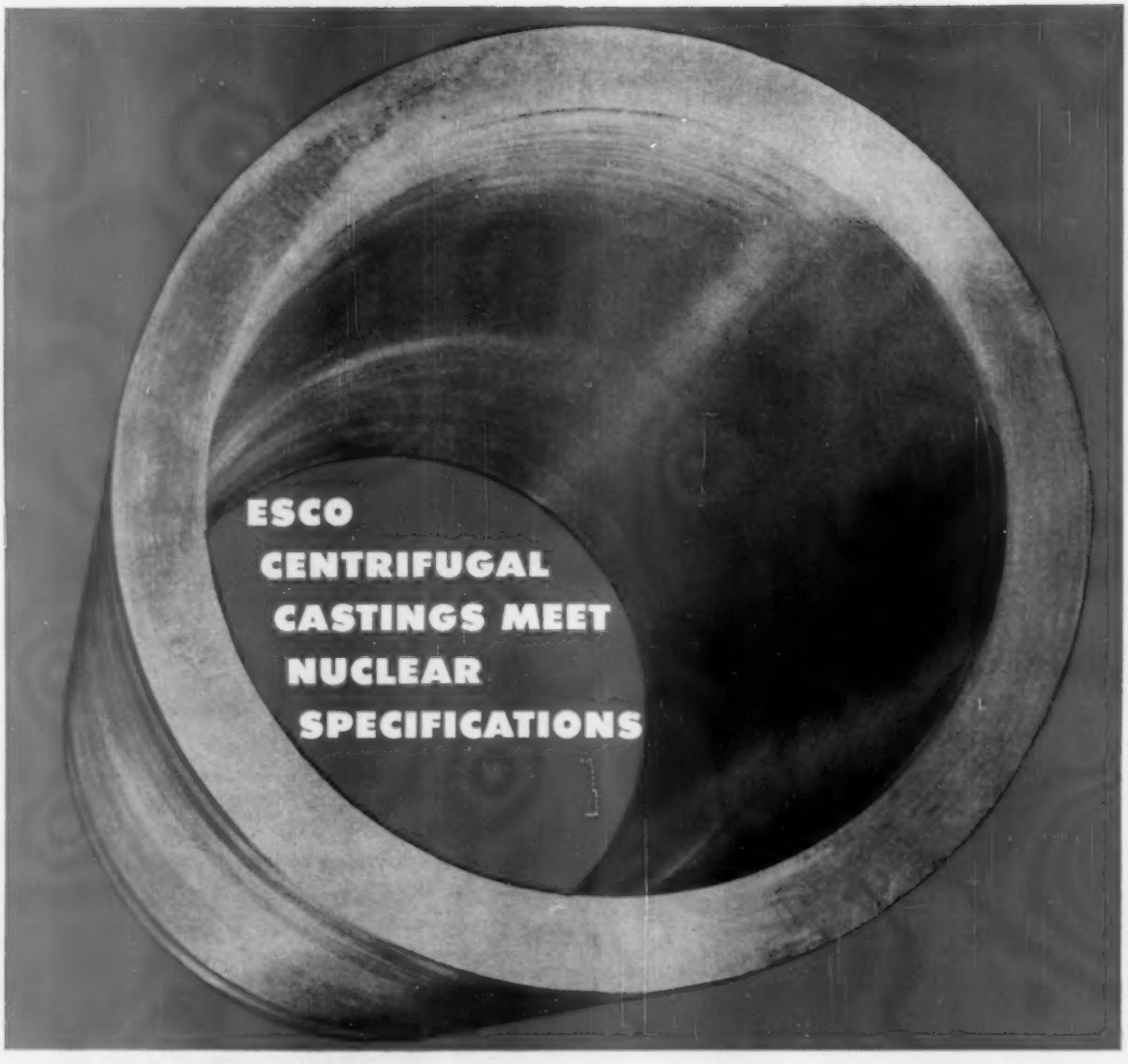

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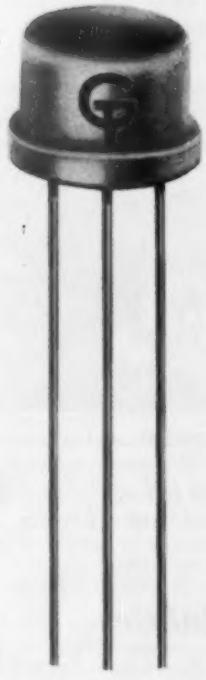

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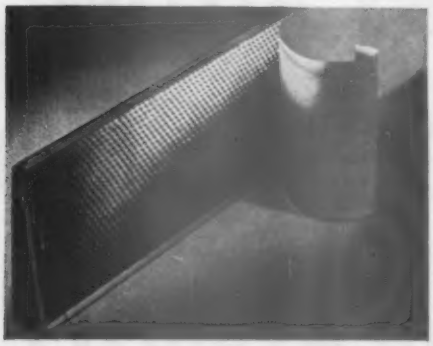

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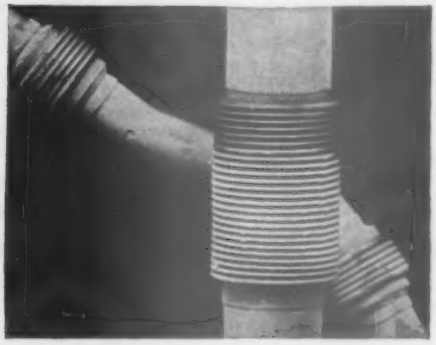

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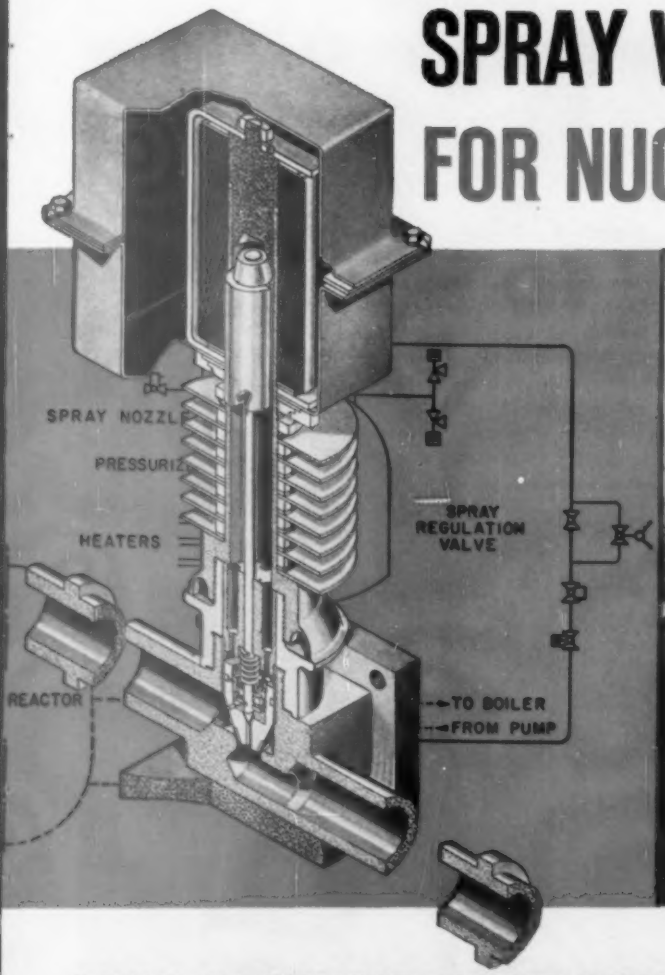

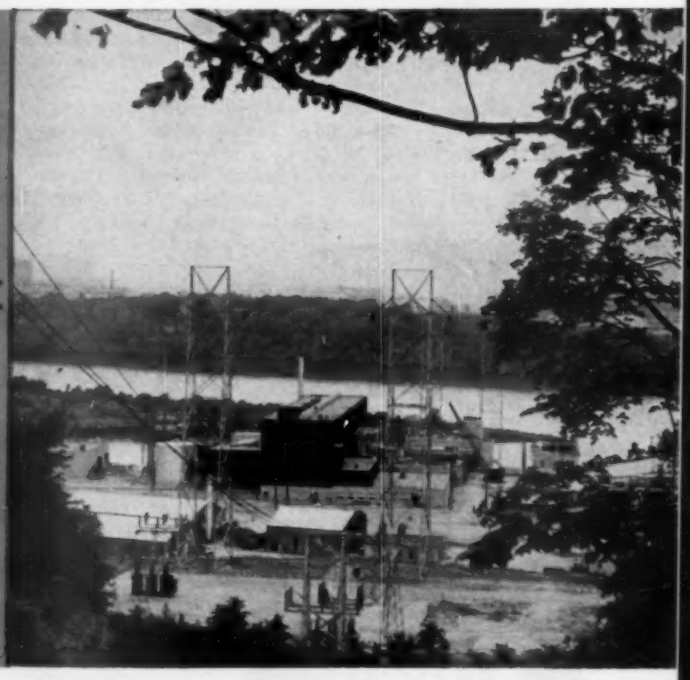

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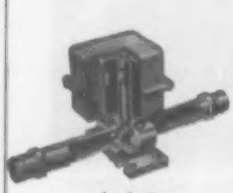

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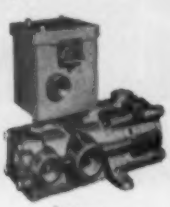

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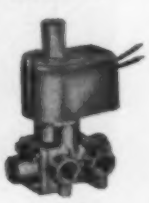

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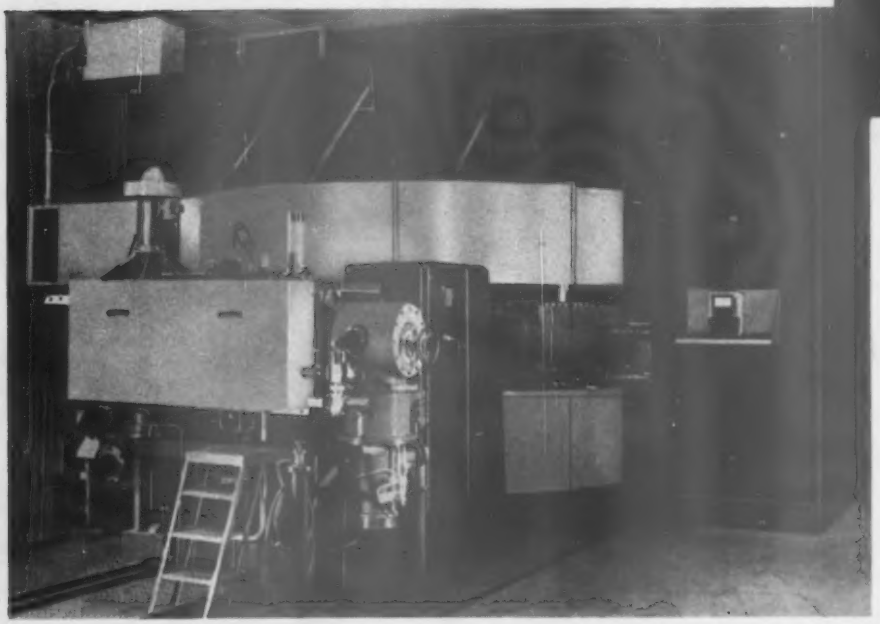

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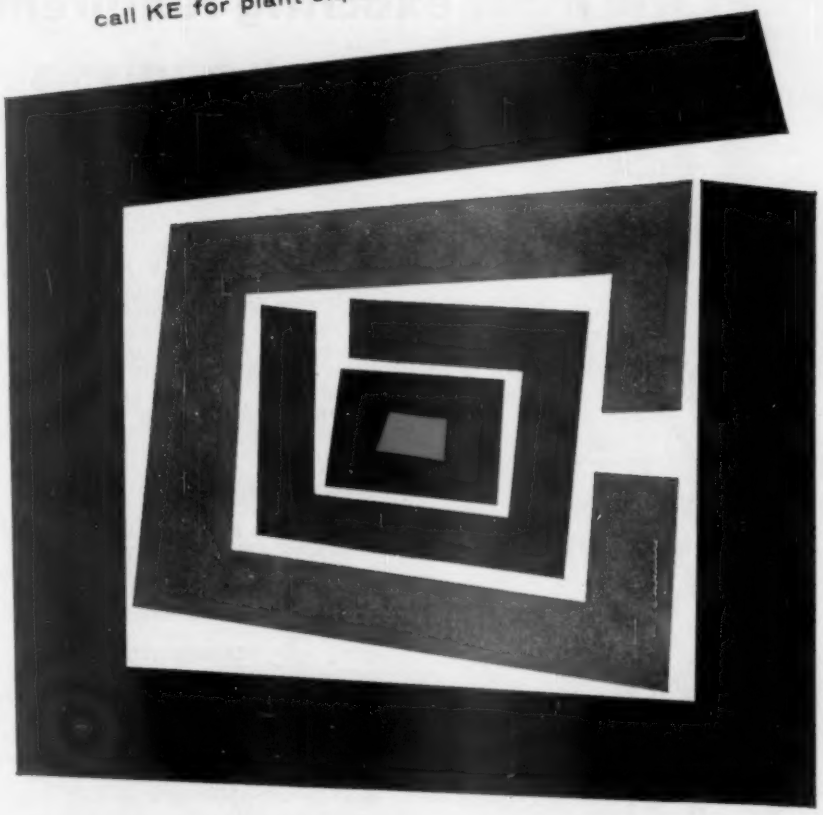

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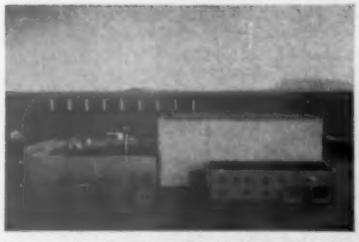

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Geneva 1958

NUCLEONICS Special Report: The Second United Nations International Conference on the Peaceful Uses of Atomic Energy, held in Geneva this month, resulted in an unprecedented outpouring of nuclear information —44 countries submitted about 2,300 papers, 650 of which were presented orally, on all aspects of nuclear energy.

Because secrecy is virtually nonexistent in peaceful applications—at least in the Western world—this second conference saw no major release of information that had been classified previously, as had been the case at the 1955 event. Despite this, there was considerable interest in the material that was presented by 13 countries on controlled thermonuclear power, a subject that was not covered in 1955, and on world experience with power reactors.

As it did in 1955, NUCLEONICS again devotes an entire issue to a report on Geneva. The following pages are based on a pre-Geneva reading of the Conference papers. Some key Russian papers were unavailable when this material was assembled. However, the Russian program and the entire Conference are covered in the on-the-spot report on pgs. 17–28.

Geneva 1958

WORLD NUCLEAR PROGRAMS

When 72 nations met at the first Geneva Conference in September, 1955, they were able to do little more than raise the question: When will nuclear fission take its place alongside coal, oil, natural gas and falling water as a basic source of energy for the generation of electricity? This month, the question was answered at Geneva:

• In the United Kingdom, Japan and most of Western Europe the present costs of electricity are high enough, the foreseeable demand large enough, or the political pressure great enough to dictate an immediate start on reactor construction and the scheduling of substantial construction programs through the decade 1960–70. Nuclear heat will be an important source of electricity in these countries by 1970.

• In the United States, West Germany, Canada and Russia—all technologically advanced nations—the availability of fossil fuels or unexploited hydroelectric sites will permit postponement of large construction programs while a broad developmental effort is made to pinpoint the most economic reactor concepts. In these areas, nuclear heat is not expected to assume basic-source status for electricity until 1970–80.

• In most—if not all—other areas of the world, a lack of demand for electricity or insufficient technological capability will not permit serious consideration of a reactor construction program before 1975. Most countries in this category, however, intend to keep themselves abreast of reactor technology so they will have the manpower and know-how to proceed when nuclear power becomes feasible (*2264*).*

Nuclear Power Economics

No nuclear power station now under construction is expected to produce competitive electricity: that is, electricity as cheap or cheaper, from a straight commercial standpoint, than that produced by a comparable conventional facility. However, paper after paper submitted to the Conference states categorically that competitive stations will be operating within a decade in some areas of the world. In consensus, the papers reflect a universal assumption that the nuclear-cost curve, moving downward, and the conventional cost curve, moving upward, will cross by early 1965 in several areas of the world where electricity consumption is substantial. By late in the decade, it is also assumed, many of the world's utility companies will be building only nuclear stations for base-load operation.

Speaking for the U. S. (*1076*), W. Kenneth Davis and U. M. Staebler predicted: "It is our considered judgment that within two or three years in several areas abroad, and within four or five years in several areas of the U. S., final design and construction can be initiated of reactor plants which confidently can be expected to be economic over their useful lifetimes. Such reactors would then be in normal operation, having passed through an initial shakedown period, by about 1964 abroad and by about 1966 in the U. S. In a few special cases economical reactors may be started even a year or two sooner."

J. C. Duckworth and E. Hywel Jones of the U. K., indicating agreement with the U. S. view, reported that the first competitive stations in that country will be completed in 1965 (*1446*). Further capital-cost reductions, they added, should take nuclear costs below the cheapest conventional plant by 1970.

Japan estimates that a Calder-Hall type reactor going into operation there in 1965 would be competitive, assuming an increase in the cost of oil during the next seven years; by 1970, reactors of this concept would be cheaper than any oil-fired station operating at 1958 prices (*1318*).

Reducing Costs

Competitive nuclear power is expected to flow from:

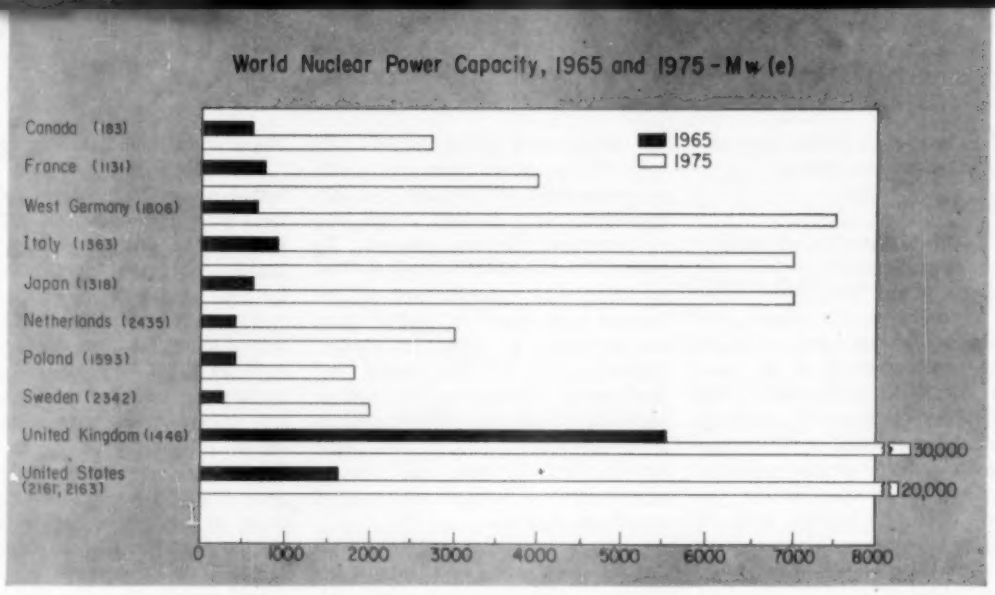
• Reductions in capital costs through the construction of more and larger plants and extension of plant lifetime.

• Reductions in fuel costs through cheaper fabrication and increases in specific power, thermal efficiency and fuel burn-up.

• Reductions in generation costs as fuel recycling and improvements in fuel reprocessing raise the value of plutonium.

The most dramatic capital-cost reductions are envisaged in the U. K.

*Italicized numbers are official Geneva Conference designations; see p. 145 for list of those references in this report.



There, costs per kw are expected to drop from £145 for the 300 Mw(e) Hunterston plant now under construction to £75 for the 1,600 Mw(e) stations scheduled for completion 1967–70. Assuming no decrease in fuel expense between the operation dates of these two plants, the drop in generation cost is expected to be from about 0.6 pence per unit to about 0.48 pence per unit, the cheapest coal-fired electricity being 0.53 pence per unit.

While estimates of capital-cost reductions are made unhesitatingly, the extent of fuel-cost reductions has been accepted as an area of too much uncertainty to permit accurate forecasts; indeed, cost increases, rather than decreases, were seen as a possibility for gas-cooled reactors as they go to larger sizes and higher temperatures. Nevertheless, many reactor builders are banking on research and development work to reduce the cost of fuel for reactors in operation by showing the way to successive core improvements. This aspect of nuclear power economics was underscored by Davis and Staebler:

"There is an unfortunate inclination to compare nuclear power plants with conventional power plants on the basis of costs at the start of operation. Actually the decisions should be made by comparing over-all economics of alternative plants over their entire estimated useful life or at least a substantial portion of it. This, incidentally, applies as well to comparisons of the relative costs of two nuclear plants—the potential for improvement over the useful life will be of great significance in these determinations. . . .

"In any conventional plant built today it can be anticipated that the cost of fuel will increase over the life of the station. In a nuclear plant, on the other hand, one may expect substantial reductions in fuel and fuel-cycle costs."

Water Storage

Accepted as the base-load power stations of the future, reactors have also come under consideration for peak-load duty in conjunction with water-storage sites.

U. K. and Swedish studies of this combination have been prompted by the possibility that reactors may be restricted to base-load operation unless schemes of this type are developed (*134, 1445*). At this stage in reactor technology it is believed that the high capital costs of nuclear plants require their operation at load factors >75%.

Salt or fresh water pumped to a storage site during the base-load operation of a reactor would be available to meet peak requirements.

"The output from these two sources of power should, at the combined load factor, provide power at a unit cost of production less than is possible any other way," according to J. M. Kay and A. A. Fulton of the U. K. (*1448*).

Heat Reactors

Reactors have been under study in both the U. S. and Western Europe as a source of industrial heat, but far less has been done in this field than in the field of nuclear power (*2342*). Recent U. S. work (*2163*) points to current cost levels for nuclear heat in the range 70–80 cents/10^6 Btu. This price is felt to be at the competitive threshold in the U. S. market and competitive already in a world market estimated to be the equivalent of 30,000 Mw of thermal capacity over the next decade.

National Reactor Programs

Canada

With an abundance of fossil fuels and water power, Canada foresees no demand for nuclear power until it can be produced for less than 6 mills/kwh (*183*). Pressurized heavy water (as moderator and coolant) and natural uranium have been employed in the two large research reactors built to date and in the first power-reactor design. D_2O has been accepted as the most efficient coolant up to now (Canada is the world's foremost advocate of D_2O) but because of its high cost ($62/kg) the Canadians have recently been looking at organic chemicals (*207, 210*), notably diphenyl and terphenyl.

The most significant reactor advance since 1955 has been the introduction of pressure tubes to replace the pressure shell (*183*). It is possible to envisage an indefinitely long life for such a station because no thick masses of metal are involved and any piece of pressure equipment may be replaced readily. Moreover, there is no obvious limit to reactor size; only "confidence founded on experience" is

needed to generate a million kilowatts of electricity from one reactor.

France

Faced with a power deficit equal to 40% of energy consumption (now met by imports of oil, etc.), France has rushed ahead with a combined construction-experimental program calling for the commissioning of one new reactor every 18 months until 1963 (*1131*). All will be natural U, graphite-moderated, gas-cooled, and each is expected to be larger and more efficient than its predecessor. France has decided it must proceed with a "crash" program because of the serious monetary burden of fuel imports and the uneasy security of Middle East oil, which makes up a larger part of its imports. Defending the "crash" approach, France's Pierre Ailleret asserted:

"Such haste is in utter opposition to normal industrial practice applied in perfecting successive prototypes; but at the same time it is justified by our legitimate concern as regards supplies of energy and by the importance of [atomic energy] to industry."

Over the next few years, the French gas reactor is expected to reach at least 250 Mw(e) in output. By mid-1965, French industry will be in a position to embark upon a large-scale program of multi-reactor power stations. The French energy picture will then be reviewed to take account of the need for a smooth transition by industry from conventional to nuclear equipment production and developments in new areas of energy supply (solar power, etc.).

West Germany

It was not until 1955 that systematic work on nuclear research and development became possible in Germany (*1806*). Furthermore, nuclear activity is restricted to peaceful utilization ("we do not consider this a disadvantage—neither at the moment nor in the long run"). Despite this late start, West Germany has operating, building, on order, or planned for order by next year eight research reactors and 4–5 power reactors. A pilot D_2O plant has been completed and two others are under study. Finally, West Germany has proposed the construction of an engineering test reactor "as soon as possible" as a Euratom project within Germany.

Nuclear power cannot compete with conventional power now but, with mounting fossil fuel imports and skyrocketing energy demand, it will play an important role from 1965 on. To prepare for this situation, five reactors of 100 Mw(e) capacity each will be ordered from abroad or built at home for operation by 1965: gas-cooled, graphite-moderated; D_2O-cooled and moderated; light-water; gas-cooled, high-temperature, pebble-bed; and organic-moderated. During this same period, Germany may build two or three reactors of foreign design without outside participation.

"This experimental program has to be considered only as a careful beginning," the German paper said. "In the not-too-distant future, however, it certainly will be followed by another more extensive and less tentative program which will provide us both with cheap power from nuclear fission and with strong sources of radiation for a wide range of useful industrial applications of different kinds."

Italy

In June 1956, the World Bank nominated Italy and Japan as the countries most likely to attain competitive nuclear power at an early date (*1363*). The Bank's subsequent arrangement with Italy to co-sponsor an international competitive bidding program for construction of a 150 Mw(e) reactor in southern Italy (SENN) pointed up Italy's barren fuel resources but ever-increasing power requirements.

The SENN project is one of five expected to give Italy 900 Mw(e) of nuclear capacity by 1965: SENN; Societa Elettronucleare Italiana—135-Mw, pressurized-water reactor; Agip Nucleare—100-Mw PWR; Societa Italiana Meridionale Energia Atomica—200-Mw Calder-Hall type; and Societa Ricerche Impianti Nucleari—two 150-Mw plants not yet selected. The SENN project is attracting worldwide attention because of its calculated effort to compare reactors now available on the world market.

Japan

Japan estimates it will add 29,300 Mw of electrical capacity to its economy by 1976—7,000 Mw in nuclear output (*1318*). To achieve this goal, Japan has worked out a 17-year, two-phase program of construction:

1. Through the first half of the period commercial stations will be of the Calder Hall type (900–1,200 Mw total). The first stations will rely heavily on imported components, but by 1967 at least 90% of component needs will be met at home. Construction of commercial reactors fueled with enriched U will be postponed pending further technological development of this concept; during the waiting period, Japan will import a 10–15 Mw(e) light-water demonstration reactor and proceed with its own development program in enriched-U technology. In the late '60's it will begin construction of enriched-U reactors at a pace estimated to equalize capacity of both natural and enriched reactors by 1975.

2. The demonstration program will also include two breeder experiments of 1–10 Mw(e) each—a thermal breeder and a fast breeder, both to be built by Japanese technicians. Japan plans to put its first commercial-size breeder in operation by 1970 [100 Mw(e)].

Japan has elected to proceed with the Calder Hall concept—to the virtual exclusion of enriched-U reactors—because of 1. the "effective operation records of the Calder Hall power plant" and 2. its "adaptability to the Japanese circumstances through its relatively easy home manufacture and acquisition of fuel." The effort on this concept will be directed, first, to "accelerating the manufacture of similar reactors (to the British design) with native techniques," then to looking for a way to achieve plutonium recycling. Eventually, Japan expects to recycle Pu produced in its own reactors and build its own breeders. Estimated foreign currency savings expected from the program outlined are $75–$107-million.

[*From Tokyo,* NUCLEONICS *has learned that since the writing of the paper outlining this program, Japan has undertaken a re-evaluation of its commitment to the Calder Hall-type reactor* (*NU, July '58, 17*).]

Netherlands

The first nuclear station [100 Mw(e)] will probably be ordered "shortly" for operation in 1962. Thereafter, construction should be accelerated to provide a capacity of 400 Mw by 1965 and 3,000 Mw by 1975. Total installed station capacity of all types by then would be 8,650 Mw (*2435*).

Sweden

This country has all of the "classic" factors underlying an ambitious reactor-construction program: the need to check a rapid growth of oil imports, large domestic deposits of U, and an advanced scientific and engineering technology (*2342*). The adopted program runs through 1965 and includes six reactors of 75–100 Mw(th) for district heating and one reactor of 100 Mw(e) for power; all are natural U, pressurized D_2O designs. Under study are three natural-U, gas-cooled concepts—two using D_2O moderators, the third using graphite. Foreseeable nuclear need for district heating purposes is estimated at 30 reactors of 30–60 Mw, 10 at 60–120 Mw and 3–4 at more than 120 Mw (all thermal); several others in the 60–120 Mw(th) range may be needed for paper mills and other industrial heat consumers.

Sweden intends to make electricity in many of its district heating reactors by adding back-pressure turbines. The first heating facility (R 3) to be so equipped will produce 15 Mw(e) and 70 Mw(th); it will be a pressurized D_2O type fueled with zircaloy-canned natural U oxide. This dual use, coupled with low-temperatures of Sweden's heat reactors (140–250° C), gives an edge on reaching competitiveness.

United Kingdom

The acknowledged world leader in terms of reactor capacity programmed is the U. K.: capacity of 1,375 Mw(e) is under construction and 5–6,000 Mw(e) are projected for completion by 1966 (*1446*). All are based on the natural U, gas-cooled, graphite-moderated technology proved out by Calder Hall. The U. K. has refused to commit significant manpower and financial resources to any other reactor concept except fast breeding, and this effort is on a long-range development schedule. The first experimental breeder reactor (Dounreay) is expected to go into operation in 1959; direction of the breeder effort beyond Dounreay will depend on what it shows and on the results of simultaneous effort to recycle Pu produced by the gas reactors.

The U. K. is also conducting its first study of a gas-cooled reactor using D_2O as moderator; the advantages would include ease of maintenance (no pressure vessel needed) and avoidance of the stored energy and oxidation problems of graphite. Construction of a reactor experiment or prototype will turn on the study now underway (*262*).

United States

The U. S. has the world's broadest technological effort and considers itself best able, in the long-run, to contribute to the world effort "to make energy really cheap" (*1076*). Broadly, the reactor concepts being explored fall into two groups: **1.** those which give promise of early attainment of economic power in high-cost areas—pressurized water, boiling water, organic-cooled and gas-cooled; and **2.** those expected to produce much lower costs or permit wider application of nuclear power—fast breeder, homogeneous, thermal breeder and others.

The U. S. program continues to be marked by a division on the issue of substantially increased government support for construction of commercial-type stations. Reactor projects scheduled for completion by 1964, each of which has some industry participation (three entirely privately-supported), total 1,300 Mw(e) (*2161*).

Nuclear Fuel Supply

A notable aspect of the three-year period since Geneva-'55—insofar as nuclear fuel is concerned—is this: it was not a period of major discovery for either uranium or thorium (*1921*). Between 1948 and 1955, U had been in total demand (for all that could be produced) by the U. S. and the U. K. The procurement policies of these two countries brought exploration and development to a high pitch and resulted in rapid expansion of production. As a consequence, the current and near-term requirements of both countries have been met from reserves discovered before 1956 and the markets to support additional exploration have not yet developed. However, as the reactor program report above shows, the market for U will broaden considerably in the 1960's and Th may come into wider commercial use later in that decade.

U reserves fall into two broad economic categories: high-grade deposits from which U oxide (U_3O_8) can be recovered and sold for $10/lb or less; and low-grade ores requiring prices 3–6 times higher for economic recovery. The leading producers of high-grade ores are Canada and South Africa (developed reserves U_3O_8 estimated at 400,000 tons each), the U. S. (220,000) and France (50–100,000). Including these and the known reserves of Australia, the Belgian Congo, Portugal and others, total reserves under development in non-Communist countries are near 1,500,000 tons/oxide.

While the period 1955–58 has been a relatively slack one for discovery of new deposits, it has been a period of tremendous growth in U production. At end-1955, total production of non-Communist countries stood at an annual rate of 11,500 tons U_3O_8. Today the rate is 35,000 tons and in a year it should reach 42,000 tons.

For the next 10–15 years it appears that a price in the range $8–$10/lb U_3O_8 should supply world requirements. With the availability of this low-cost material the prospect of having to turn to high-cost sources (uraniferous shale and phosphate) seems much farther in the future than it did a few years ago. However, the low-grade deposits, if they are needed, are sufficient to assure adequate supply for many generations.

Thorium

Known reserves of Th are about half those of U (*862*) and some 500,000 tons of ThO_2 are believed to be economically recoverable. The major portion of these is in the beach placer sands of India and the Blind River and Bancroft U districts of Canada; the rest lies in placer and vein deposits in Australia, Brazil, Madagascar, South Africa, southeast Asia and the U. S.

A major development since Geneva-'55 has been the discovery of Th in association with the Blind River and Bancroft U deposits—about one part ThO_2 is available for two parts U_3O_8. Moreover, approximately 85% of the ThO_2 is dissolved during the processing of U and is readily available for inexpensive recovery after extraction of U by ion exchange.

Geneva 1958

FUSION

Thermonuclear power, which was only alluded to at the 1955 Geneva Conference (in Homi Bhabha's opening speech), came into its own at Geneva-1958, where it was by all counts the center of attention and occupied six full sessions. Those sessions revealed that man still has much to learn about the fully ionized state of matter. All the national programs (see descriptions in this section) are plagued by problems of plasma instability, impurities from walls and the like. No nation has even produced thermonuclear neutrons with certainty, much less reached the temperatures (100,000,000° in the case of the DD reaction) at which inevitable radiation losses can be matched by fusion power produced.

U. S. Leads in Diversity of Approach

The U. S. controlled-fusion research program encompasses four major and two minor experimental directions. Achievements have been impressive (some of the highlights are detailed on following pages) but much more remains to be done. At Geneva, Edward Teller summarized (*2410*) where the program stands with respect to its ultimate goal: the heating of a mass of deuterium gas to temperatures of about 100,000,000° K and its confinement for periods of time sufficient to permit extraction of more energy than was required to heat the gas in the first place.

With regard to temperatures: most of the present devices have reached "temperatures" of few million degrees. The record is >10,000,000° K in the mirror machine (*377*) followed by the stellarator and the pinch devices. (Direct comparison isn't fair—the pinch devices operate at densities that are a factor of ten or a hundred higher.)

As to confinement times—here the criterion is more complicated: for

Oak Ridge's DCX

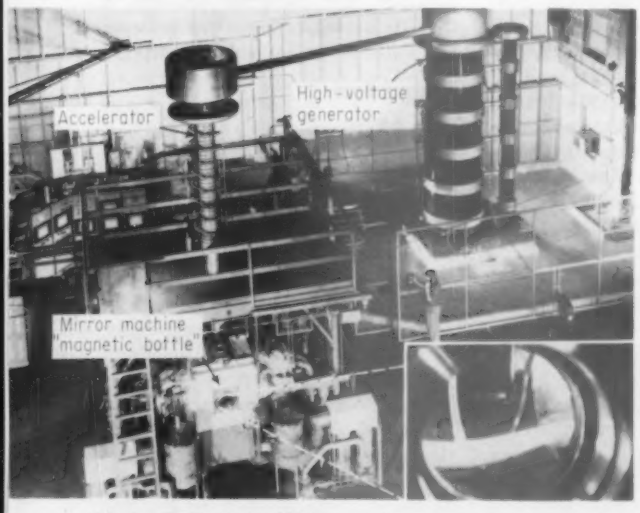


FIG. 1. DCX includes 600-kev accelerator that propels molecular deuterium ions into special carbon-arc discharge (see inset) where many of them are dissociated, forming 300-kev atomic ions that can be trapped in magnetic bottle and eventually produce hot plasma

Livermore's Pyrotron

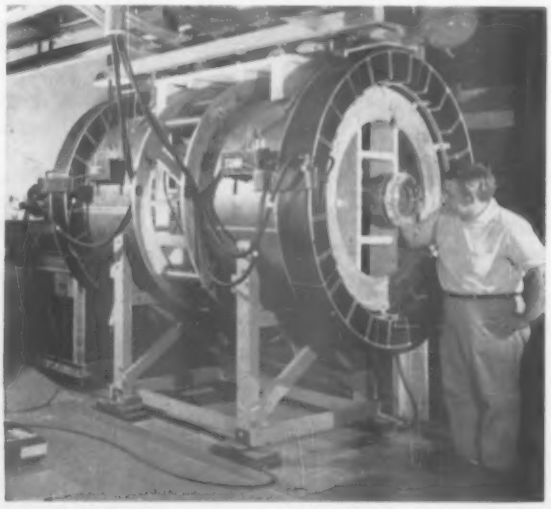


FIG. 2. Pyrotron, or mirror machine, uses axial magnetic field that is stronger at ends than in center to form a two-necked "magnetic bottle." Plasma can be injected through these necks and then adiabatically compressed and heated by increasing magnetic field strength

steady-state devices one wants essentially infinite confinement; for pulsed devices one wants the product of the confinement time τ (in sec) by the particle density n (in cm^{-3}) to be of the order of 10^{16} for the DD reaction. Pinch-effect-plasmas have $n \simeq 10^{17}$ and have confinement times of 10^{-6} sec, therefore $n\tau \simeq 10^{11}$; for the Livermore mirror machine $n \simeq 10^{13}$ and $\tau \simeq 10^{-2}$, so again $n\tau \simeq 10^{11}$. Thus we are a factor of at least 10^5 from the goal.

Speaking loosely, then, we are closer to achieving the necessary temperatures than the necessary confinement times. Actually confinement is the basic problem in any case, for if the confinement problem were solved one could take time with the heating process and probably accomplish it in whatever way was necessary. As it is, only partial confinement has been achieved and success in achieving higher temperatures depends in good part on the elimination of instabilities such as plague thermonuclear devices the world around. No one can say for sure if sufficient stability can be achieved by any of the present Sherwood devices—but the very diversity of the U. S. program adds hope that this can in fact be done by one approach if not by another.

U. S. Fusion Program

Here are the highlights of Project Sherwood as they were revealed in Geneva papers:

Oak Ridge—DCX

A new and very special kind of carbon-arc discharge lies at the heart of Oak Ridge's thermonuclear experiment, which is primarily a plasma-injection method. First discovered by J. S. Luce in 1955 and kept secret until Geneva (*1790*), the arc has two very valuable properties: (a) It readily dissociates molecular ions that pass through it. (b) It can operate without gas feed in vacuua down to 10^{-7} mm and lower. In fact the arc serves as an ion pump—as used in DCX it has a pumping speed of 4,000 liter/sec and as such is the most effective pump on the main chamber.

The arcs have electron energies of tens to hundreds of electron volts and require an axial magnetic field for stable operation. Arcs as long as 6 ft have been drawn at 150 v and 250 amp. The arcs are still only partly understood, but their efficacy is attested by results with DCX.

The Oak Ridge thermonuclear experiment (*544*) involves the injection of 600-kev deuterium molecular ions into a "magnetic-mirror" region in which is located a Luce discharge (Fig. 1). The 300-kev atomic ions produced by dissociation are trapped in a closed circular orbit near the central plane of the machine, while the resultant neutral atoms and the remaining undissociated beam leave the machine at once. Since the initial vacuum in DCX is 10^{-6}–10^{-5} mm, the trapped beam remains highly organized —each deuteron loses very little energy before being lost by charge exchange with a neutral gas atom (pertinent cross sections are 2×10^{-15} cm² for charge exchange and $<10^{-19}$ cm² for energy loss).

However, the ionization cross section is sufficiently high ($\sim 4 \times 10^{-17}$ cm²) that about 20 neutrals are ionized by each hot ion in the trapped beam before it is neutralized and leaves the system. If the rate of ionization of neutral gas inside the plasma can be made greater than the rate at which neutrals enter, "burnout" will take place—the neutral density will decrease rapidly and, with increased current, the ion density will rise until the plasma is totally ionized. For a spherical plasma of radius equal to that of 300-kev ions in a 10-kilogauss field, burnout occurs at 80 ma (atomic ion current) for 10^{-6} mm background pressure and at 0.8 ma for 10^{-8} mm pressure.

Once burnout has been achieved, charge exchange will no longer be the dominant mechanism and energy will be shared with colder ions and electrons by means of Coulomb and, finally, neutron-producing nuclear collisions. The neutron production should be substantial: for 1 ma of dissociated deuterons entering into a spherical plasma volume of the size specified above, the ion density would be $\sim 10^{13}$/cm³ and the total neutron yield $\sim 10^{13}$/sec—which would be unmistakable evidence of the formation of a hot thermonuclear plasma.

The Oak Ridge group seems to be very close to achieving burnout—trapped ring currents of 4 amp have been measured, and with a background pressure of 5×10^{-7} and

Princeton's Stellarator

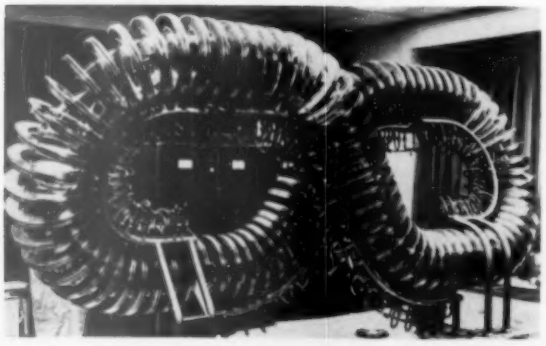
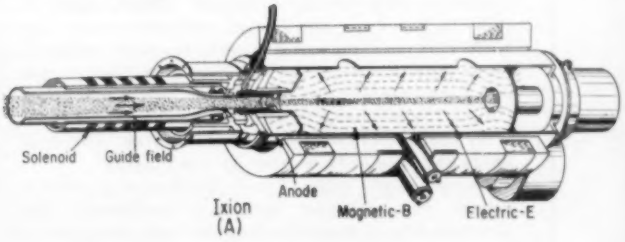


FIG. 3. Stellarator's figure-eight geometry is obvious in this photograph of model developed for Geneva exhibit. Shape is designed to overcome drift of ions to the walls; similar effect can be achieved with special windings on a simple torus

New Pinch Devices



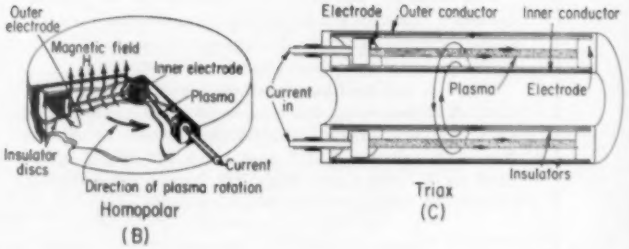


FIG. 4. Ixion (A), Homopolar (B) and Triax (C) are new pinch devices. First two use crossed electric and magnetic fields to induce rotation of the entire plasma. Triax is designed to explore stability properties of sheet-like plasmas

a 40-μamp dissociated beam they are within a factor of ten (less pressure or more current) or so in light hydrogen, which is what they have been using preliminarily.

They also have prospects for related devices involving deuterium energetic arcs operated within coaxial hollow gas arcs.

Livermore—Pyrotron

The basic containment principle of the pyrotron is that charged particles tend to be reflected from regions where magnetic field lines converge (*377*). For example a system of two coils (as shown in Fig. 2) will form an axial magnetic field that "necks in" at the position of the coils and therefore tends to confine a plasma in the space between coils. This confinement is not perfect; in particular particles traveling in helices with small pitch angles to the axis of the machine can pass through the mirrors [the critical angle is given by $\sin^{-1}(H_O/H_M)^{\frac{1}{2}}$ where H_M and H_O are the magnetic field strengths at the peak of the mirrors and the lowest value in between the mirrors, respectively].

Interparticle collisions can shove particles onto escape trajectories; thus confinement is better at higher temperatures, where the collision rate is lower. In particular if the mean particle energy is 150 kev and the particle density 10^{14}/cm³, the confinement time is of the order of 0.5 sec, sufficient for net power production. (One way to improve confinement time further might be to rotate the plasma; see Ixion in Fig. 4.) An interesting and important sidelight is that impurities tend to be confined less well, thus easing what is normally a critical problem in thermonuclear devices.

Heating in a mirror machine can be accomplished by radial or axial compression of the plasma—accomplished by increasing the field strengths or moving the mirrors together, respectively. It is also possible to transfer the plasma from one region to another by moving the mirrors.

Injection of the plasma into a mirror machine can be accomplished in several ways; injection through the mirrors into a rapidly rising magnetic field and use of an ion source located within the confinement volume are two such methods.

In one experiment an initial plasma burst filling the confinement volume to a density of 10^{11}–10^{12}/cm³ was stably compressed to a final density of 10^{14}/cm³ by using magnetic compression ratios of 1,000:1 in times of about 500 μsec. Confinement times of the order of 1 msec were observed, with energetic ions remaining for as long as 20 msec. Electron temperatures were 10–20 kev; ion temperatures are computed to be about 1 kev, 10,000,000° K. No instabilities were observed but the particle densities are so low as to prohibit the conclusion that power-producing mirror machines would also be stable.

Princeton—Stellarator

The stellarator, like Livermore's mirror machine, uses an axial magnetic field to contain its plasma, but in toroidal rather than linear geometry (*2170*). However, since ions would drift to the wall in a simple torus, the doughnut is twisted into a pretzel-like figure eight (see Fig. 3). Even greater stability can be achieved by using a simple torus with supplementary helical windings (*1875*). (This is the arrangement to be employed for the large-scale Model-C Stellarator under construction at Princeton.)

Initial heating in a stellarator (to temperatures of ~$10^{6°}$ K) is to be done by making it the secondary of a transformer through whose primary a large current from a capacitor bank is caused to flow. This is similar to the initial heating in a toroidal pinch device but the currents are much lower—in fact it is necessary for stellarator stability that the current remain below a certain value (dependent on magnetic field) at all times. Because the collision cross section decreases rapidly with energy some of electrons can gain energy continuously and "run away" from the main Maxwellian distribution (*358*). This makes theoretical heating calculations somewhat uncertain, and further experiments will be needed to prove the feasibility of ohmic heating.

Subsequent heating is to be by so-called "magnetic pumping"—radiofrequency oscillation of the confining magnetic field "pumps" the plasma and heats it (to thermonuclear temperatures, in principle). The frequency can be chosen to match one of several natural periods of the device. Unfortunately it has not yet been possible to verify this type of heating because of high plasma losses due to poor confinement and impurities.

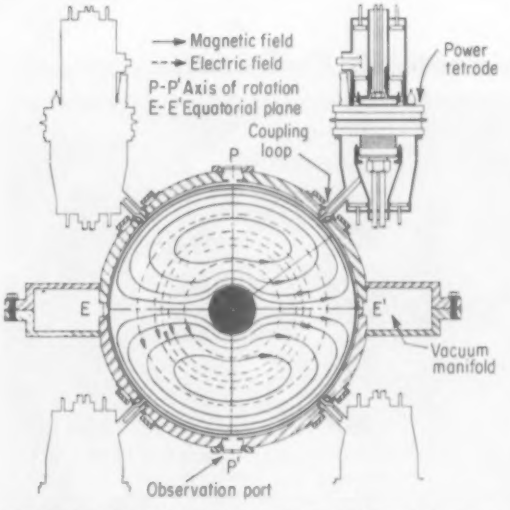
This matter of impurities in outgassed from the walls of a stellarator is of utmost mportance since stellarators operate at pressures well below those in pinch devices

Argonne Gets Into the Act

Argonne National Laboratory has joined the roster of organizations engaged in the quest for controlled thermonuclear power. Their approach (*350*) is based on the use of radiofrequency (10–1,000 Mc) electromagnetic fields for both containment and heating (see figure). The idea is not new, yet this is a departure from the pinch effect (where the discharge is a damped pulse rather than truly oscillatory) or even the Stellarator (where r-f fields are used only to heat up the plasma).

As one might expect, such a different approach brings new promise and new problems. The promise derives from the fact that the structure of r-f fields is fundamentally different from that of static fields (the curl is nonzero) thus making novel field configurations possible. Also with an oscillatory confining field there is opportunity to take advantage of possible dynamic stabilization effects—and stability has been the bugaboo of all plasma devices. Finally r-f fields can (in principle) simultaneously perform the functions of containment, heating and direct electrical power extraction.

Biggest, and perhaps fatal, defect of the scheme is that skin losses in the resonant cavity associated with the scheme far exceed the power produced (for cavity shown wall losses are 5,200 Mw if operated at a level such that TD reaction produced only 0.324 Mw!). Work is at an early stage.



PROPOSED THERMONUCLEAR EXPERIMENT would use 8 beam-power tetrodes (4 not shown) to provide rotating 800-Mc electromagnetic field configuration in 0.3-meter spherical cavity to contain plasma at density of 2×10^{12}/cm³ ($p \cong 10^{-5}$ mm, rather lower than other fusion devices) and heat it to temperature of 1 kev. Unfortunately skin-effect losses in cavity walls absorb almost all of tubes' 1-Mw (total) output

($\sim$1 μ is opposed to $\sim$100 μ). Considerable effort has gone into ultra-high-vacuum research, and all recent stellarators have been designed baked at 400° C. Initial pressures of 10^{-10} mm have been achieved.

Another very important tool in the battle against impurities is the "Divertor," a device for averting contact between the hot plasma and the wall of the discharge tube. An outer cylindrical shell of the magnetic flux is arranged to pass into an auxiliary chamber where outward-diffusing ions can strike a collector plate. Reductions in impurity level of 2–3 have been achieved; much higher values should be obtainable when improved confinement around the machine insures that ions don't leak out before they reach the divertor section.

Devices. Typical of the stellarators built thus far is the B-65 machine (*360*), which is in the form of a racetrack 207 in. long with a helical winding. A divertor is provided on one leg, an r-f heating chamber on the other. The vacuum chamber is 0.083-in.-wall 4-in.-diameter stainless-steel tubing. Hollow water-cooled $\frac{3}{16}$-in.-diameter copper wire insulated with 12 mils of Mylar wrapped on the surface of the tube carries the helical field current. The same wire, in a second layer, forms the solenoidal confining-field winding. The vacuum system is unbaked and the machine base pressure is 10^{-6} mm. A 13-kv 0.6-Megajoule capacitor bank energizes the magnetic field, which rises to its maximum in 5.5 msec. Iron- and ferrite-core transformers provide, respectively, a unidirectional 0.1 volt-sec ohmic-heating electric field and a 250-kc r.f. breakdown electric field. In one experiment, an applied ohmic heating field of 0.24 volts/cm was used to give approximately 2,000 amp of plasma current. The helical windings limited the diameter of the main discharge column to about 2 in.

LASL—Pinch and Beyond

Following its stated (*1860*) aim ("to steer clear of any large machine, keeping to a course of fairly basic plasma physics research in which theory and practice are intimately associated, around numerous modest scale experiments") Los Alamos has come up with an impressive list of new devices—Ixion, Scylla, P.A.M.—to supplement the already unveiled Columbus (linear) and Perhapsatron (toroidal) pinch machines.

Pinch research. The spectrum of neutrons from the high-powered linear Columbus II (NU, Feb. '58, 90) shows shifts due to ordered pinch motion, as if a segment of the plasma, within which the deuterons are colliding randomly, moves with appropriate velocity towards the cathode. The lower-power linear Columbus S-4 has been used for fundamental observations, which have proved reproducible (for stable phenomena) to within a few per cent. In particular, magnetic-probe measurements have made possible accurate measurements of axial and azimuthal components of magnetic field whence can be derived plasma pressure (nkT) distribution.

The origins of instabilities in this pinch can thus be studied more carefully and fitted in with new theoretical studies such as those by B. R. Suydam. Suydam's results indicate that undirectional axial magnetic fields will not suffice to stabilize the pinch, and that it may well be necessary to reverse B_z in the vacuum region outside the pinch with respect to its swept-in axial field.

The latest in the toroidal Perhapsatron series, Perhapsatron S-4, has 7-cm bore, 35-cm major diameter, is made of quartz. Fed at two points by two 225-μf condensers charged to 20 kv the new apparatus will generate 320-kiloamp peak pinch currents in 12.5 μsec. Up to 10^7 neutrons per pulse have been observed, but they are associated with instabilities. Further results must await the return of the apparatus from the technical exhibit at Geneva. Meanwhile a still larger S-5 is under construction.

Los Alamos has also done research on large-bore metal-wall systems akin to those favored by the British. They use a 6-meter-long linear tube of 15 cm diameter with 1.6-cm-thick Al walls (Columbus T-1) and a 3.6-meter long 90-cm-diameter stainless-steel welled Columbus T-2 bellows. One offshoot is a proposal for a continuous-metal-spiral pinch tube wall (*2348*) whose high inductance would keep it from shorting out the discharge and might even further aid stability by providing the reversed magnetic field that seems desirable by Suydam's criterion.

Scylla. Another Los Alamos device is Scylla (*356*) a linear device in which a fast pulse through appropriate single-turn windings simultaneously produces a radially converging shock wave and the mirror fields to contain the plasma it produces. Up to 2×10^7 neutrons are produced on the second and third and fourth current maxima of the current, which has the form of a damped oscillation of 200-kc frequency. Thus far there is nothing to indicate that these neutrons are not truly thermonuclear, in which case the plasma temperature would be 13×10^{6}° C.

Ixion. The spinning-plasma device, Ixion III, (see Fig. 4a) is essentially a mirror machine to which has been added a radial electric field. As a consequence the containment "tightness" of the magnetic bottle is improved (the centrifugal force of the plasma rotation induced by the crossed electric and magnetic fields keeps the ions away from the axis where they might more readily pass through the "necks" of the mirrored bottle). There is good evidence to indicate that a substantial amount of the energy fed into the device ends up in bulk rotation of the plasma; in this the machine is closely related to Berkeley's Homopolar device (see p. 67). Impurities seem to be limiting the attainable energy and therefore future efforts will be concentrated on reducing these.

P.A.M. injectors. High plasma velocities have been achieved with a plasma injection device (*355*) that depends on the immediate ionization before dispersal of a puff of cold gas injected into an evacuated tube provided with an axial magnetic field. Initial ionization is either by pinch or by tangential electric field from a single-turn coil that is connected to a high-voltage condenser. Any neutral gas ahead of the slug of plasma is either ionized and carried along or left behind, so that the plasma can be observed thereafter in vacuum. The plasma has been further accelerated by a traveling magnetic wave injected into a solenoidal transmission line—velocities of 5×10^6 cm/sec and impulses of 90 gm cm/sec have been achieved with order of magnitude improvements in the offing. Clearly such a device might have potential application to propulsion problems (NU, July '58, 73).

Berkeley, Livermore Pinches

Rotating plasmas, sheet plasmas, "screw-dynamic" pinches are among the new de-

Continued on p. 144.

Industry's Fusion Role

Geneva provided a clearer picture of the role being played by U. S. industry in fusion projects lying outside AEC's Project Sherwood.

Pinch-effect experiments (*352*) were revealed to be part of the privately financed program being conducted by General Atomic and a group of Texas utilities (NU, June '57, 17). Emphasis has been on strong axial stabilizing magnetic fields (1–20 kilogauss) in aluminum-oxide discharge tubes of 4.4 and 6.8 cm i.d. and 30-cm length. Typical of the fundamental approach that has characterized General Atomic's researches, papers were also presented on hydrogen-atom collision cross sections measured with an atomic beam apparatus and on magnetohydrodynamic theory.

Vitro Engineering Co. reported an engineering evaluation of a schematic 1,000-Mw (net) D-D reactor (*2358*). They postulate a steady-state system utilizing a 50-kilogauss solenoidal confining field 350 ft long. A pressurized-water blanket picking up both neutrons and bremstrahlung energy is the primary heat-transfer system. The inner radius of the heat-transfer blanket they choose so that the bremstrahlung heat flux does not exceed 5×10^5 Btu/hr/ft^2. The magnet o.d. is 18 ft. A surprising result is that maximum power generation occurs at the lowest permissible operating temperature ($\sim$33 kev). Capital cost comes out $550/kw(e)—but this is a highly preliminary figure.

U. K. Steadfastly Pursues the Pinch

"Whilst other approaches to the problem of producing and maintaining a very high temperature gas will undoubtedly be described to this Conference, research in the United Kingdom has been devoted to a study of gas confinement by the 'pinch effect,' that is confinement by the magnetic field which accompanies an axial flow of current through the gas. Many other possible methods have been studied at earlier times but none appeared so promising."

This statement by P. C. Thonemann (*78*), who heads up the controlled fusion effort at Harwell, provides a capsule description of the British position in the quest for thermonuclear power.

Eventually this singlemindedness may pay dividends (as it has in the fission field with gas-cooled reactors) but at Geneva the effect was to make the British story less impressive than that of the U. S. This was particularly true since the British had revealed major aspects of their program last January (NU, Feb. '58, p. 90). Thus Britain's Geneva Conference contribution in the fusion field consisted primarily of a restatement and putting into perspective of the already-disclosed efforts at the three major sites—Atomic Energy Research Establishment, Harwell (*1519*), Atomic Weapons Research Establishment, Aldermaston (*1460*), and the Associated Electrical Industries laboratory, also at Aldermaston (*3*). In addition there was a review of diagnostic techniques (*1520*) and several theoretical contributions (*1, 2, 33*). Some of the highlights are summarized below.

Fast Pinch at AWRE

A condenser bank with lower inductance (for its size) than any before is providing rates of current rise up to 2×10^{12} amp/sec for fast linear pinch experiments at the Atomic Weapons Research Establishment (*1460*). The essential feature of the bank is the use of a large number of separately switched condensers. Such shock heating of plasmas may have advantages over ohmic heating, which becomes inefficient at high temperatures. The low inductance of the system leads to several striking differences from earlier Russian and U. S. fast-pinch results—for example, the neutron production is lower and not substantially reduced by axial magnetic field stabilization. This suggests that the detailed behavior during early stages of formation of the pinch discharge may have greater effect on its behavior than has been generally appreciated.

Slow Pinch—AERE, AEI

Experience with Zeta (*1519*) and Sceptre (*3*), which had been discussed in January (NU, Feb. '58, 90), was updated at Geneva and presented in greater perspective. Neutrons observed in Zeta have been shown to be nonthermonuclear (NU, June '58, 19). Torus liners are being improved and the capacitor bank enlarged prior to the next round of experiments.

AEI—Wall Materials

The British have put a lot of effort into the study of materials for the segmented metal walls of their pinch devices, largely because they were forced to by the problem of arcing and sputtering of impurities into the discharge. An interesting paper (*34*) summarizes their experience and extrapolates it to possible full-scale machines. For the latter situation the high rate of soft X-ray and fast-neutron irradiation (100 watts/cm^2 and 7×10^{14} n/cm^2/sec, respectively) makes the wall problem a critical one, particularly since any impurities entering the discharge will increase the bremstrahlung rate, which, at best, is still $\frac{1}{6}$ the power-production rate. Metal walls of low atomic number look best, with beryllium a particularly good bet, except for embrittlement due to the production of He under neutron irradiation. The future course of fusion-reactor development may intensify or lessen wall problems, but to some extent they will always be present.

Russians Concentrate on the Pinch, Too

As best as can be gaged by the skimpy abstracts and three complete papers (*2228, 2300, 2301*) available at this writing, the status of the Russian fusion effort seems to be as follows:

- They have devoted most of their fusion efforts to the pinch effect.
- Their experimental techniques are good and, in some cases, novel.
- They have not achieved thermonuclear neutrons nor reached temperatures higher than those reached in the West.
- They have done theoretical studies of a Stellarator-like machine, and of other non-pinch devices.
- They are particularly strong in the theory of collective effects, which could prove very significant since these are the kinds of effects that seem to be bedeviling the U. S. program at the present time.
- Apparently the Russians have not attempted to alter the basic pinch geometry or magnetic field configuration substantially, but have sought to attain a deeper experimental and theoretical understanding of the phenomena that it affords. One finds no discussion of rotating plasmas or sheet pinch devices as in the U. S.

For a final accounting on the Russian fusion performance at Geneva, readers are referred to the cabled on-the-scene report, p. 17.

Fast-Pinch Studies

Using a series of ceramic and metal pinch tubes (*2301*) at voltages of 20–120 kv and maximum currents of 200–1,600 kiloamp from a 0.5-Megajoule bank with rates of current rise of 10^{11}–10^{12} amp/sec, the Russians have conducted a long series of linear pinch experiments. Neutrons are observed and shown to be nonthermonuclear by nuclear emulsion studies of their spectra.

Piezoelectric Probes

One experimental detail where the Russians offered something new is in plasma diagnostics. They use piezoelectric transducers pressure to supplement the magnetic pickups (also used by U. S. researchers) that measure plasma pressures by inferring them from measurements of magnetic field strength.

Typically these piezoelectric probes consist of a small (0.5-mm-dia) piece of piezo-ceramic at the end of a porcelain rod used for positioning. They are calibrated again at a diaphragm-type transducer in the range of working pressures (0–100 atm).

Smaller Countries Show Interest in Fusion

That thermonuclear research is not the exclusive province of great powers is attested by solid contributions from several smaller nations including Sweden (*147, 1708*), Switzerland (*1471*), Germany (*1056*), Poland, Japan (*1329*), Rumania, France, Italy, Denmark, and Turkey. Here are some of the highlights from their Geneva papers:

Swedes Pinch, Simulate

Toroidal pinch-effect experiments at Uppsala (*147*), underway for the past two years, have been bolstered by the recent addition of a large toroidal device (Fig. 6) powered by an impressive 1 Megajoule condenser bank (8,200 μf at 15 kv). The condenser bank is in 40 separate groups each containing 24 condensers; each group can feed one of a corresponding set of 40 pairs of coils around the torus—one set can generate a longitudinal magnetic field, the other can induce a longitudinal electric field. The coils can be powered in various combinations for maximum experimental flexibility. Peak coil currents can be as high as 1.6×10^6 amp. The discharge tube itself has 130 cm major diameter, 28 cm minor diameter.

Another Swedish contribution is a study of a pinch-effect analog—a freely falling vertical stream of mercury replaces the gas discharge thus providing a more easily observed medium (*1708*). Voltage is applied along the axis of the 4-mm-diameter 14-cm-long stream as shown in Fig. 7. The entire apparatus can be placed in a uniform magnetic field. Sausage and kink instabilities have been observed. Unfortunately stabilization by a longitudinal magnetic field cannot be demonstrated since the field lines "escape" from the liquid more readily than in the case of the plasma discharge.

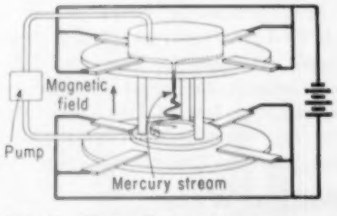


FIG. 7. Pinch-effect analog uses mercury stream to simulate plasma. Extreme kink-type instability is shown

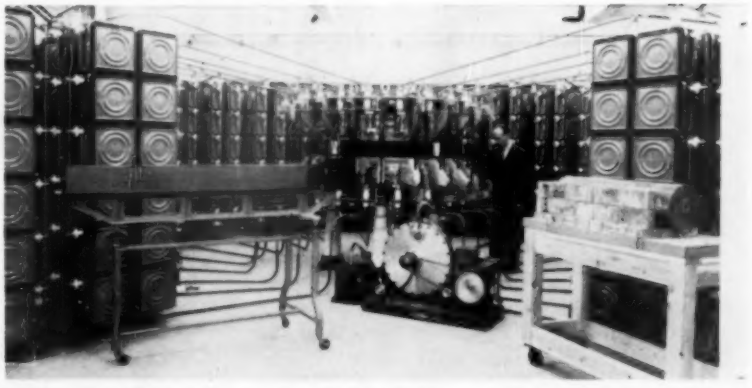


FIG. 6. Toroidal pinch apparatus in Kai Siegbahn's laboratory in Uppsala, Sweden is surrounded by 1-Megajoule condenser bank

Japanese Innovate

Instead of the usual pinch-effect experiments the Japanese propose to work with axial-magnetic-field devices (*1329*). As a variant on this they propose to start with a static magnetic field parallel to the axis between two oppositely rotating slotted conducting discs. Each disc acts as the rotor of a homopolar generator and a gas discharge started between the edges of the disc is pinched into a tube at the center. Thus one has an attempt at conversion of mechanical energy to plasma energy. In another proposal, resembling the American Astron device being developed at Livermore (*2446*), a beam of relativistic electrons is to be used to ionize, heat and confine low-pressure deuterium.

German Program

The extensive German program was reviewed in a single paper presented by L. Biermann of Göttingen (*1056*). Linear pinch experiments have been conducted at Munich, Stuttgart and Aachen. Toroidal pinch experiments have been conducted only at the Max Planck Institute for Physics, Göttingen. In addition the Germans have conducted extensive theoretical work on heating methods, stability problems and other aspects of plasma physics.

Swiss Study Blankets

Tritium breeding in a lithium blanket surrounding a fusion reactor has been the subject of calculations (*1471*) by Hinterman and Widerõe of Brown Boveri. The breeding reaction is, of course, $Li^6(n,t)He^4$, whose cross section has a resonance at 255 kev and is $1/v$ below. Neutrons from the fusion reactions—of energy 2.5 Mev from the DD reaction and 14 Mev from the DT reaction—must be slowed down before they can be absorbed efficiently. Because of its low scattering cross section, a blanket of natural lithium would have to be at least 100 cm thick to do this; therefore the Swiss team proposes using a graphite or beryllium moderator lined with lithium on both sides. The idea is that the moderator will do essentially all of the slowing down but the lithium is there to capture the thermalized neutrons (mean free path for absorption of thermal neutrons is 0.3 cm in natural lithium and 0.023 cm in Li^6). This also increases the concentration of tritium by reducing the quantity of lithium.

Assuming that all the bred tritium is recycled back to the reactor, it is computed that for a plasma operating at a temperature of 200 million degrees (20 kev) and with a blanket that produced 0.72 tritium atoms per neutron, only 1.21% of the atoms in the discharge would need to be tritons, the remainder being deuterons. Thus the initial tritium investment need not be prohibitive. Even then, because the DT cross section is 100 times the DD and because a triton is produced in half of the DD reactions, the DT reaction will occur three times as frequently.

Geneva 1958

REACTOR PROJECTS

If Geneva-'55 was the conference of promise, insofar as nuclear power is concerned, Geneva-'58 is certainly the conference of early results. Three years ago in the dewy period of hopeful beginnings, only three cases of actual operating experience with power reactors were presented—two by the U. S., one by the U. S. S. R. 23 other power reactors in the design stage were described. This year, a still optimistic but wiser and more sophisticated audience heard recitals of 24 cases of operating experience with power reactors, descriptions of 46 others under construction in 12 countries. They saw the U. S. show performance comparison charts by the dozen of a variety of reactor types (selection opposite), heard the British describe three generations and four avenues of development of one type. No longer could there be any doubt that nuclear power—whether economic, or not yet—was well on its way to becoming commonplace.

U. S. Operating and Design Experience

A variety of reactor types has been operated at temperatures permitting efficient conversion of nuclear energy to electricity (*1075*). Water, sodium, air and organic materials (terphenyls) have proved successful coolants. Fuel has been used both in the form of solid elements and as a fluid. All degrees of neutron moderation—thermal, intermediate, and fast unmoderated—have been tested and stable operation demonstrated. The temperatures explored ranged from 418° F to >1580°.

Pressurized Water

The pressurized-water reactor has enjoyed the longest and widest use in the U. S. to date. Some specific experience:

Submarine Thermal Reactor (STR). Development of pressurized-water power reactor technology began in 1951 with construction of STR, land-based prototype of the power plant for the Navy't USS *Nautilus*. STR first produced substantial power on May 31, 1953, and has been refueled twice.

Pressurized water power plants have operated satisfactorily in the submarines *Nautilus* and *Skate*. By April 30, 1958, *Nautilus* had steamed over 107,000 mi, 74,000 of them submerged; *Skate* had logged over 12,000 mi, 10,000 submerged.

Shippingport Atomic Power Station (PWR) has been described by its builders (*2462*) as "a facility for exploring the performance of a pressurized-water reactor as a function of time," its flexibility permitting a wide range of experiments under operating conditions. Completed less than one month behind schedule, the plant has done duty delivering 60 Mw(e) under commercial conditions—a power level higher than that of any operating reactor in the world today.

"The plant was placed in operation without a failure or modification of significance and has performed essentially as designed" (*2462*). The plant is stable in operation, primarily due to the self-regulating characteristics associated with its high negative temperature coefficient of reactivity. This was dramatically demonstrated during the first week of operation when, while at full power, the turbine was tripped out; the reactor followed down to almost zero output without scramming, without any control rod movement, without any relief valves lifting (NU, April '58, 70). Operational experience indicates that the measured flow, flow distribution and pressure drop across the core are very close to design values. Also, power sharing between seed and blanket, as well as coolant temperature rise in each of the orifice regions, is in close agreement with predictions. At present ~50% of the heat produced is generated in the blanket; with time, more and more power will be derived from plutonium in the blanket.

Several problems have shown up in recent months' operations. One is an oscillation in the power distribution resulting in an unequal power distribution of 3–5%, with a period of ~24 hr during normal operation, with this value rising to ±20% following burnout of xenon from the peak of the Xe transient. Experience has shown this imbalance can easily be corrected by control rod positioning. Another problem: the thermocouples intended to read temperature at the center of the seed fuel plates have not performed successfully, and a new seed cluster with revised instrumentation is being prepared for insertion. The failed element detection and location system has operated well; it has detected small defects in one of the blanket assemblies that have resulted in a minor amount of fission products in the coolant. The amount, very small, is within tolerance.

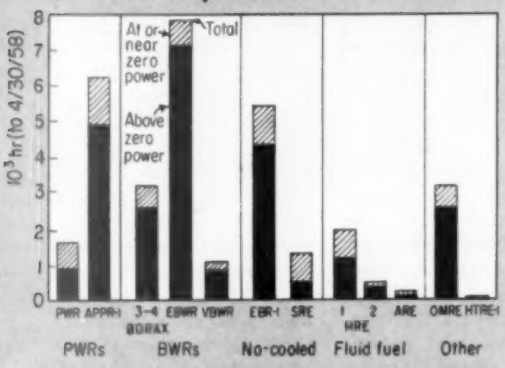
Army Package Power Reactor (APPR-1). Performance has been highly satisfactory. This reactor has also demonstrated great stability even at complete loss of load (NU, May '57, 26). By March 1, 1958, the plant had generated 7,500 Mwh of electricity.

Boiling Water

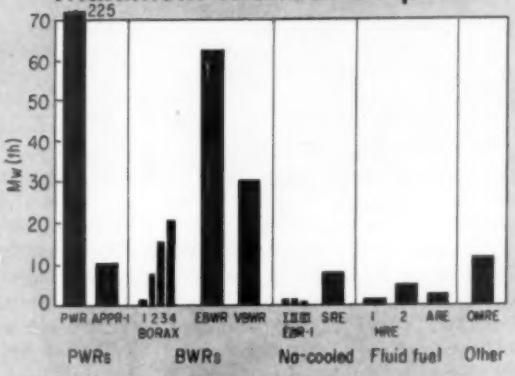
Boiling water reactors have also shown very satisfactory stability; under

Characteristics and Performance of U.S. Power Reactors

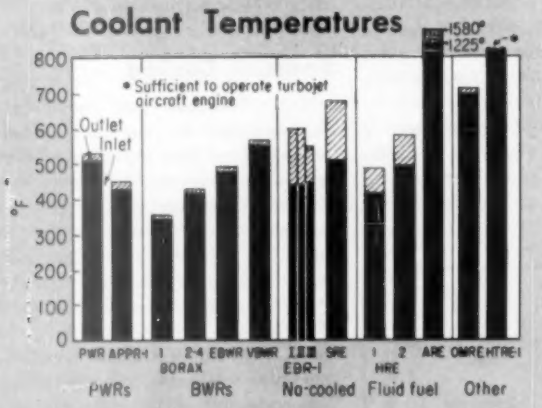
Nuclear Operation


10^3 hr (to 4/30/58)
At or near zero power
Above zero power
Total
PWR APPR-1 3-4 BORAX EBWR VBWR EBR-1 SRE 1 2 HRE ARE OMRE HTRE-1
PWRs BWRs Na-cooled Fluid fuel Other

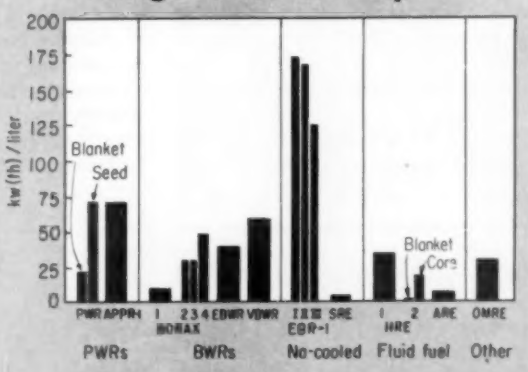
Maximum Thermal Output



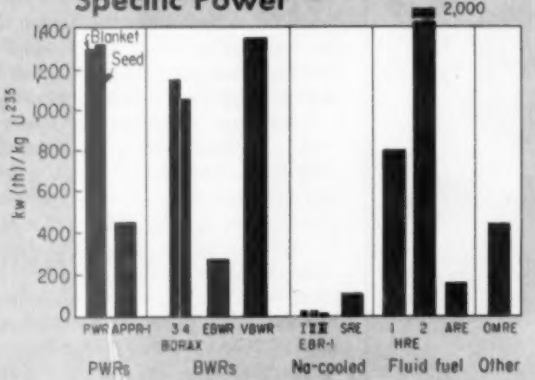
Coolant Temperatures


°F
* Sufficient to operate turbojet aircraft engine
Outlet
Inlet
1580°
1225°
PWR APPR-1 1 2-4 BORAX EBWR VBWR III EBR-1 SRE 1 2 HRE ARE OMRE HTRE-1
PWRs BWRs Na-cooled Fluid fuel Other

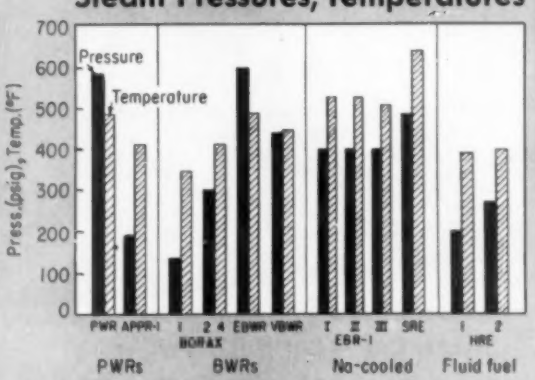
Average Power Density


kw (th) / liter
Blanket
Seed
Blanket
Core
PWR APPR-1 1 2 3 4 BORAX EBWR VBWR III EBR-1 SRE 1 2 HRE ARE OMRE
PWRs BWRs Na-cooled Fluid fuel Other

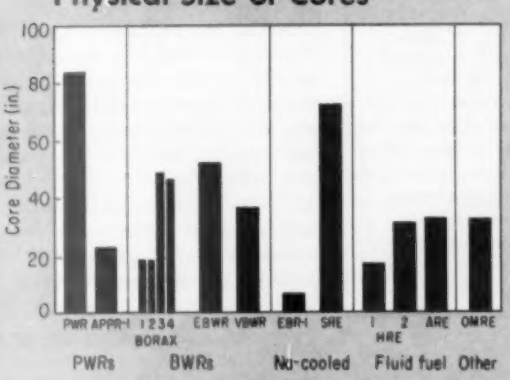
Specific Power


kw (th) / kg U^{235}
Blanket
Seed
2,000
PWR APPR-1 3 4 BORAX EBWR VBWR III EBR-1 SRE 1 2 HRE ARE OMRE
PWRs BWRs Na-cooled Fluid fuel Other

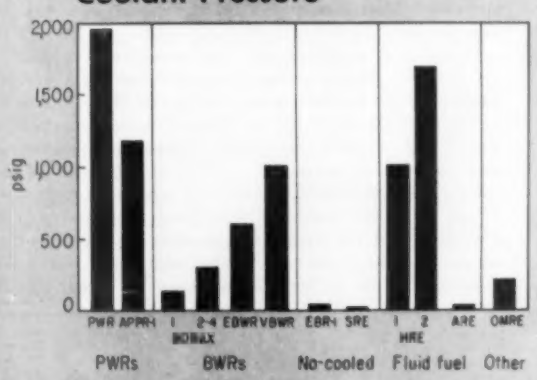
Steam Pressures, Temperatures


Press. (psig), Temp. (°F)
Pressure
Temperature
PWR APPR-1 1 2 4 BORAX EBWR VBWR I II III EBR-1 SRE 1 2 HRE
PWRs BWRs Na-cooled Fluid fuel

Physical Size of Cores



Coolant Pressure

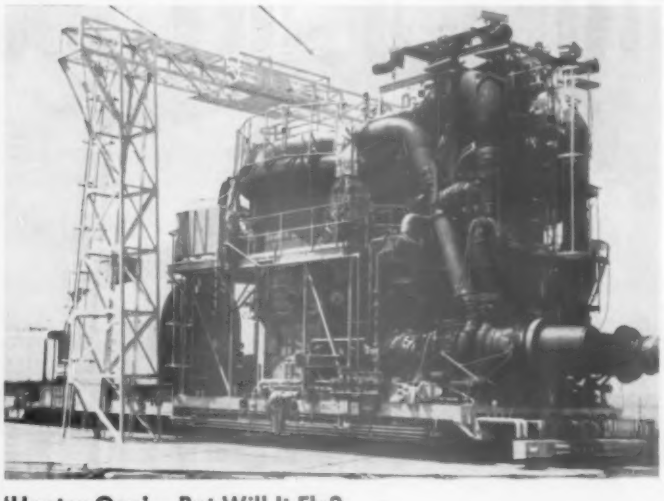


normal operating conditions they have a large margin against reactivity instability caused by variations in moderator density. Turbine radioactivity after extended operation has proved negligible (NU, June '58, 25).

Experimental Boiling Water Reactor (EBWR) has successfully operated at a power density of 65 kw/l at 600 psig (*2379*). The reactor has not reached a power-density limit because of limitations in associated plant. Designed for 20 Mw(th), it has run at 62 and stopped there only because feedwater pump capacity ran out. Reliability of power output was shown by absence of accidental or nonscheduled shutdowns during routine operation. Stability at 600 psig and at power levels of 20–62 Mw has been remarkably good, even with the entire bank of nine control rods inserted more than halfway to hold down reactivity when no Xe is present. Response to imposed changes in ambient pressure and in feedwater flow has been as expected. Radioactivity levels in power plant equipment have been acceptably low; no significant increase was noted even during operation with a deliberately-faulted zirconium-clad element.

The presence of a small amount of fission-product gases has been observed in the exhaust from the main condenser air ejectors since start of operation. Quantitative evaluation showed that these could result from surface contamination of fuel elements totaling only a few milligrams of uranium. This, together with the constant magnitude of the fission-product activity, has ruled out the possibility of fuel-cladding failure. The experience shows that monitoring these discharge gases is a very sensitive method for detecting fuel-cladding failures.

Vallecitos Boiling Water Reactor (VBWR) has been operated at 900–1,000 psi at all power outputs to 20 Mw(th) with no signs of instability. Principally because of its high operating pressure, VBWR could be operated at average power densities throughout the core of >50 kw/l with natural circulation (*1923*). The reactor responds very smoothly to load changes, follows turbine output from zero to full in <1 min with no severe drop in pressure. At 50 Mw(th), radiation levels in the turbine building are <6 mr/hr, except in the immediate vicinity of the condenser and the turbine's high-pressure stage, where levels reach 15 mr/hr. Contact levels as high as 100 mr/hr have been measured on the air ejector. Experience shows turbine-building personnel receive <30 mr/wk in performance of their duties.



'Heater One'—But Will It Fly?

One of the freshest revelations about the U. S. Aircraft Nuclear Propulsion program at Geneva (*465*) concerns the first Heat Transfer Reactor Experiment (HTRE-1), the direct-air-cycle reactor that first powered an aircraft turbojet engine (NU, Aug. '56, R1). HTRE-1 is air-cooled, water-moderated, metallic-U fueled. Shield is water, steel, lead. Three series of operational tests were run: 400 Mwh in January 1956 including runs of 6 and 80 hr; 1,600 Mwh in May and June; 3,300 Mwh with a new set of fuel elements in October-December. "The reactor was always stable and responsive to both manual and automatic control. . . . The third series of tests was successfully conducted at the predicted temperature. At the completion of 100 hr of operation, it was decided to increase the reactor temperature and continue in operation in excess of the power-plant minimum requirements. The additional operating period lasted a total of 40 hr under nuclear power." That was HTRE's last run.

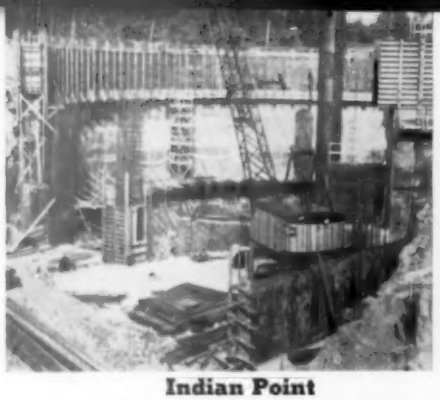
Sodium Cooled

Both thermal- and fast-neutron sodium-cooled reactors have operated successfully. Thermal shock problems have been encountered. Maintenance of primary sodium systems has been possible without great difficulty. No long-lived radioactivity has been noted to be deposited in primary components.

Experimental Breeder Reactor (EBR-1) produced the world's first nuclear electricity back in Dec. 1951, and was described in Geneva-'55 paper 813. Reporting on events since the fuel meltdown in late 1955 (NU, June '56, 35), it was stated that the bowing-proof elements of the Mark 3 core installed late in 1957 have eliminated the earlier instability.

Submarine Intermediate Reactor (SIR). A land-based prototype for USS *Seawolf* operated for two years, 1955–57, but suffered leaks in both superheaters and one steam generator. It was deactivated when the Navy decided against further shipboard sodium reactors. However an essential duplicate aboard *Seawolf*, after similar pretrial leak problems, has performed without a hitch; by April 30, 1958, *Seawolf* had steamed over 40,000 mi, 27,500 of them submerged.

Sodium Reactor Experiment (SRE). Three power runs were carried out late in 1957 (*452*). Previously, information on temperature distributions near the main tank nozzles indicated advisability of restricting the temperature differential across the reactor to 150° F until means for improving the match between reactor power and coolant flow after scram could be installed. For this reason, these runs were made at ⅓ of full power. After



Indian Point

Fermi

Dresden

AUGUST 1958—This was state of construction at three major U. S. power reactor sites

installation of an eddy-current brake in March 1958, full power was achieved in May. The sodium piping has operated at 750° F with no evidence of leaks. Oxide plugging has been experienced in some of the 1½-in. and 1-in. sodium lines; it is expected to clear up when all lines are filled and oxide content of the bulk sodium is reduced to <10 ppm. In the secondary system, when oxide content has been kept below 10 ppm, no plugging has been observed.

Leakage has been observed in seven bellows-sealed valves. One corrected itself. As far as can be determined, failure was for operational reasons rather than inherent weaknesses. The three conduction pumps initially used for sodium transfer tended to plug at the throat; they have been replaced with linear induction pumps, which have given 3,200 hr of trouble-free operation. Preparation for the power runs and physics test operations provided about 700 fuel-element transfers with the fuel-handling cask. Time for one element change is 2 hr.

Other Types

Fluid Fuel. Homogeneous Reactor Experiment 1 (HRE-1) circulated liquid for 4,500 hr, 2,000 of them critical, with uranyl sulfate, during its two-year life 1952–54. It demonstrated an extremely high degree of inherent stability, not even needing the control rods provided. HRE-2, a larger successor, after early troubles (NU, Jan. '57, R4; Nov., 22; Feb. '58, 26; May, 26), is now being operated as a single-region system and is performing satisfactorily (NU, Aug. '58, 21). The Aircraft Reactor Experiment (ARE), a molten-uranium-salt fueled system, operated briefly in 1954 at maximum steady-state temperature of 1,580° F and power levels up to 2.5 Mw(th) (NU, June '57, 20). Los Alamos Power Reactor Experiment 1 (LAPRE-1) was abandoned due to failure of gold cladding, needed to contain uranyl phosphate; work on LAPRE-2, a variant, is continuing.

Organic Moderated. Although OMRE's steady-state operating program has barely started (*421*), by Jan. 1958 it had accumulated 80 Mwd power operation at bulk coolant temperatures of 500–700° F and fuel-plate surface temperatures up to 750° F. Samples of coolant from the main loop have been taken on a routine basis and analyzed for high boiler residue content: any material less volatile than para-terphenyl (see Fig. 1). Preliminary value is 3.8 lb of high boiler residue formed per Mwh operation. Accessibility of primary components for maintenance is possible even during operation. Highest radiation level in the process area was 80 mr/hr at surface of the 10-in. pipes and of the filter removing particulate matter from the system; level in the center of the area was 13 mr/hr, less than twice normal tolerance. Airborne activity was always well below breathing tolerance.

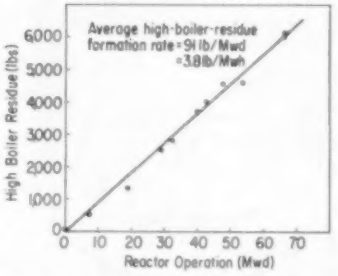


FIG. 1. OMRE: PLOT of weight of high boiler residue formed as a function of Mwd of reactor operation. Points shown represent data obtained under widely-varying temperatures—all, however, in range of interest for power reactor operation

Under Construction

Five large-scale commercial nuclear power plants are actually under construction in the U. S. at the present time (*2447*) and in varying stages of advancement (*see photos*). They are the Indian Point, Yankee, Dresden, Fermi and Sheldon stations. A smaller plant at Elk River is just getting under way, and seven more are in preparation.* Thus the AEC's varied development program is being followed by private industry with two pressurized water and four boiling water units, one sodium-graphite, one fast breeder, one homogeneous; two heavy-water pressure-tube designs, an organic moderated and a sodium-deuterium type.

At the Indian Point site where Consolidated Edison of N. Y. and Babcock & Wilcox are building for the former's grid a 275-Mw thorium-converting pressurized-water plant (*1885*), erection of the cylindrical concrete envelope for the 160-ft-dia steel reactor sphere is well along. Initial operation is set for late 1960. Power cost from the $90-million plant is placed at 13.2 mills/kwh.

Westinghouse's second-generation large pressurized-water plant for the

* Two boiling reactors of 66 and 60 Mw(e) respectively by Allis Chalmers for Northern States Power Co. and General Electric for Pacific Gas & Electric; a 17-Mw heavy-water cooled and moderated pressure-tube unit by Westinghouse for the Carolinas-Virginia group and a high-temperature, gas-cooled D_2O-moderated pressure-tube design by General Nuclear Engineering for the East Central and Florida West Coast nuclear groups; a 70–150-Mw single-region homogeneous reactor by Westinghouse for Pennsylvania Power & Light Co.; a 12.5-Mw Atomics International organic-moderated unit for Piqua, Ohio, and a 10-Mw sodium-deuterium reactor by NDA for Chugach Electric Cooperative of Alaska.

Yankee group is proceeding at the Rowe, Mass., site (*1038*). Research and development is 60% complete; the 125-ft-dia reactor sphere and fuel element production are both starting this fall. Cost of power from the 134-Mw, $50-million unit is conservatively put at 14 mills/kwh.

Fifty mi SW of Chicago, General Electric is building on a turn-key basis a >$45-million, 180-Mw boiling-water reactor: now about 60% completed, it is unique in being six months ahead of schedule (*2372*). Power costs will be competitive, 7.5 mills/kwh, for Commonwealth Edison on the basis of its $30-million investment, which does not include contributions of GE and six associated utilities.

The Enrico Fermi Atomic Power Plant, a 100-Mw fast breeder being built 35 mi SW of Detroit by Power Reactor Development Co. and Detroit Edison (*1850, 2427*) is well advanced with the 92-ton reactor vessel installed in the containment dome. Construction is to be essentially completed in February, when a year-long nonnuclear testing program will start; meanwhile the fuel elements have been ordered. $20-million has been spent on research and development.

Site work is now also underway at Hallam, Neb., for the 75-Mw sodium graphite reactor for the Sheldon station (*608*). Completion is set for Jan. 1962, and full-power operation in June.

Also described were four AEC power reactors: the Argonne Low Power Reactor (*455*), which went critical last month, is a small package plant supplying 300 kw(e) and 400 kw(th) space heat with minimum supervision. In the construction phase are the Experimental Breeder Reactor-2 (*1782*) and Plutonium Recycle Test Reactor (*447*), while LMFRE-1 is still under development (*2355*).

Operating and Design Experience Abroad

Calder Hall

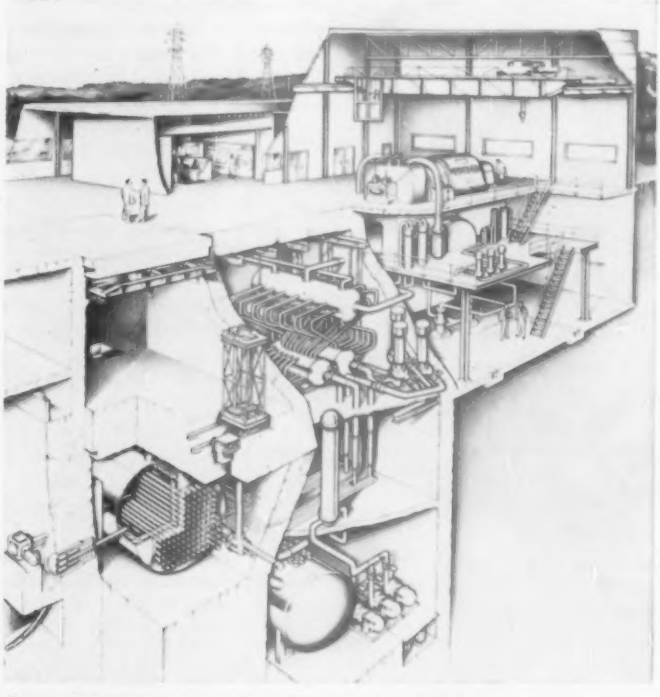
During 18 months ending March 1958, two of the four reactor units that make up the Calder Hall station have generated 690,000,000 kwh of electricity (*1522*). This represents a full-power load factor of 70% on Reactor No. 1 and 78% on No. 2.

Plant startup. In the initial startup from the cold condition, 24 hr are required to achieve the required reactor inlet temperature. Under normal conditions, when the plant is shut down, the coolant circulators and reactor inlet ducts are maintained at operational temperature (~140° C) using either fission-product heating or injection of low-pressure steam from an external source into the heat exchangers.

The 48 boron-steel control rods are normally operated in line and may be withdrawn at only one speed (providing $\sim 2 \times 10^{-4}$ k/sec) so that 4 hr is required before criticality can be attained. On reaching criticality, coolant flow is maintained at the minimum controllable value of ~15% of maximum, and the fuel surface temperature is raised to the operating maximum value of 408° C at the present arbitrarily-chosen rate of 2° C/min by continuing control-rod withdrawal. During this phase the reactor gas outlet temperature rises to the final operating value of ~330° C, and the steam generated in the heat exchangers is dumped.

At this point (~2 hr after criticality) the turbogenerators are run up and synchronized with the national grid supply. They are operated on steam-pressure control so that generation will automatically follow changes in reactor heat output. The dump condenser is left available with its inlet valves set at slightly higher pressures so that it will take over the steam load if a turbine is tripped out.

From this stage the plant is operated at constant temperatures, and power is adjusted by variation in coolant flow. To raise power, coolant flow is increased. Rates of increase of power of

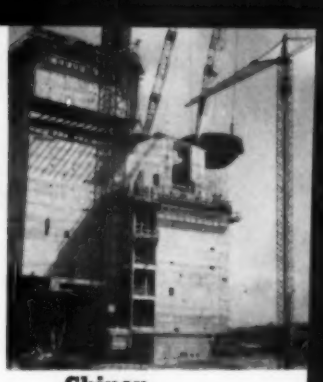


Canada's Pressure-Tube Design for NPD

The first Canadian power reactor has been redesigned to make use of the pressure-tube concept (*209*), and work has resumed at the site (NU, July '58, 25). The barrel-shaped drum containing horizontal pipes at lower left is the reactor; below it further left is fuel-storage bay, and below it to right is moderator dump tank. In between, extending to right of reactor and hanging from crane is fuelling machine. Vertical tank to right is moderator cooler, and above that is steam generator. 20-Mw(e) plant is to be completed in 1961.

Bradwell

Berkeley



Chinon

AUGUST 1958—This was state of construction at two British and a French power reactor site

up to 15 Mw(th)/min can be used, but the normal operational rate is 5 Mw(th)/min to match the turbine loading rate of 1 Mw(e)/min, which allows power to increase in 30 min from the minimum flow condition of 30 Mw(th) to the full power condition of 180 Mw(th).

Normal operation. The reactor control points are the reactor gas inlet temperature, maintained at 140° C, and the maximum fuel-element surface temperature, maintained at present at the probably conservative value of 408° C. The reactor gas inlet temperature is maintained constant by adjusting the low-pressure circuit steam pressure in the range 30–60 psi. A change in pressure of 4 psi changes the gas inlet temperature by 1° C, but the system is sluggish: gas temperatures follow pressure adjustment with a delay of 5–10 min. Thus, the system is suitable only for trimming under steady conditions and cannot be used for correcting any rapid changes of temperature that can be caused, for example, by variations in feed water flow.

The maximum fuel-element surface temperature is achieved by control rod movements. Initially about 20 fuel-element temperature measurements were provided to cover the 1,696 fuel element channels. This was found to be completely inadequate. About 60 temperature-measuring points are now provided, which is about the limit with the present design, and even these do not give a complete temperature picture. Extrapolation from actual measurements is frequently required, and, because of possible variations of the order of 15° C caused by fuel-element and coolant-channel manufacturing tolerances and flux distortions, conservative interpretation of the measured values is necessary.

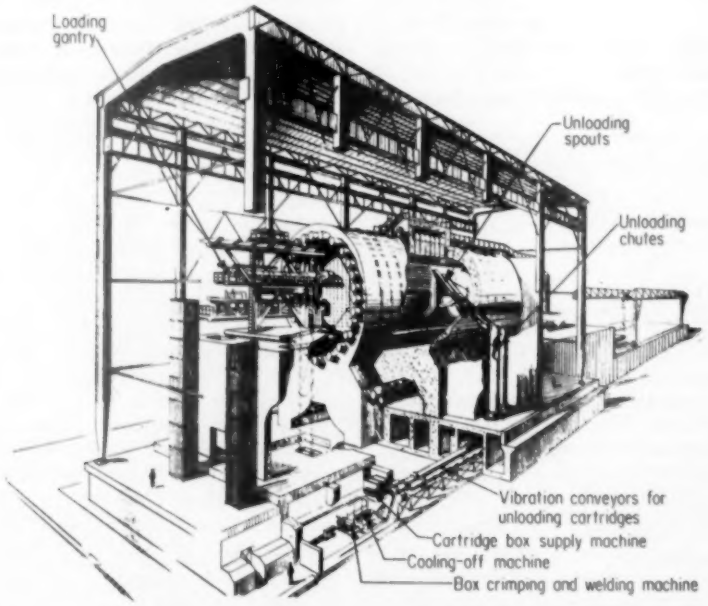
After startup, the control rods must be withdrawn over the first three days to compensate for the 1.6% in reactivity lost by the buildup of xenon. The total reactivity available has been adjusted (by loading fixed absorbers in fuel-element channels) so that the control rods are almost fully withdrawn at the completion of Xe buildup.

The reactors have a high plutonium conversion factor (>0.8), and the reactivity changes after the first three days of operation are due mainly to the buildup of samarium and plutonium at the rate of ~0.2% per month. After about three months of operation, the control rods are inserted in the core nearly half way.

Control rods can be inserted into any of 112 positions in the reactor. The initial pattern consisted of 40 rods closely grouped around the center. With this pattern, rod insertion for reactivity buildup caused flux peaking just outside the control-rod region. With fuel elements in this region at maximum permitted temperatures, the remaining fuel elements were running cold, and the reactor would not operate at a power level of more than 170 Mw(th). Consequently the control-rod pattern was changed to provide a more uniform distribution over the whole core, the number being increased to 48 to maintain the same amount of control, and the coolant flow pattern was improved so that power now can be maintained at levels above 200 Mw(th) for periods of 3–4 months.

The performance of the feedwater-flow regulators has a marked effect on the reactor. The rate of feedwater

Continued on p. 130.



FRANCE'S G-2, sketched to show apparatus for loading and unloading while in operation

Geneva 1958

NEW REACTOR CONCEPTS

Britain Plans Advanced Reactor Concepts

Research effort by the British on advanced gas-cooled reactor concepts is proceeding along two different levels. The more immediate objective (*312*) is to develop a slightly enriched gas-cooled system using UO_2 and Be cladding that will give at least 25% better economic performance than the Calder Hall generation of reactors and that will lend itself to Pu recycling. The longer range goal (*314*) is to produce a high-temperature Th-U^{233} breeder with a gas outlet temperature around 750° C. So far the first program includes plans to build a 100 Mw(th) experimental reactor; the second program has produced a design for a 10 Mw experiment that features graphite-clad fuel elements with provisions for continuous fission-product removal.

Advanced GCR

To prove out the slightly enriched UO_2-Be concept the Atomic Energy Authority is proceeding with the design and construction of a 100 Mw(th) experimental advanced gas-cooled reactor (the AGR project). The plant will be completed early in 1961. Engineering parameters are shown in the table.

The Be-clad UO_2 fuel will operate at 4–5 times present fuel ratings and at temperatures 200° C above Calder values. The reactor will be flexible enough to accommodate cores that differ radically in design. It will also be large enough so that there will be sufficient channels to get adequate statistics on fuel element performance and (with an unflattened core power distribution) there will be a large number of very hot channels for advanced testing.

The heat exchangers and main gas circulators will be flexible enough to handle a smaller number of channels at higher average heat rating and the same total power. Each fuel channel has individual access through the pressure vessel head and is equipped with its own instrumentation and flow control. Refueling can be carried out with the reactor on load and at full operating gas pressure. The design arrangement would allow the graphite moderator to be completely replaced or reconstructed.

Advanced Breeder

Plans for Britain's ultimate power reactor concept are pointed towards a gas-cooled system that will:

- have an outlet temperature high enough to operate modern steam turbine generating sets or, if possible, a fairly efficient gas turbine cycle.
- have high fuel burnup
- be able to breed with thorium

The AEA has chosen 750° C as the outlet temperature goal; this figure is a little higher than the minimum needed for modern steam temperatures (600° C) and also would permit the use of a gas-turbine driven circulator (see box, p. 80).

Core materials. An average outlet temperature of 750° C means that maximum fuel surface temperatures may approach 1,000° C or more. Although stainless steel and zirconium are usable as cladding at 1,000° C, the former would absorb too many neutrons and the latter has high-temperature chemical compatibility problems. Moreover to get a reason-

Design Data for Advanced Gas-Cooled Reactor

Reactor heat rate	100 Mw(th)
Net electrical output	28 Mw(e)
Core length	14 ft
Core diameter	15 ft
Number of channels	250
Gas inlet pressure	270 psig
Gas inlet temperature	250–325° C
Gas outlet temperature	500–575° C
Maximum fuel element surface temp.	600° C
UO_2 fuel	14.6 tonne
Mean reactor rating	7.75 Mw/tonne U
Peak reactor rating	18.00 Mw/tonne U
Initial enrichment	1.65 C_0 (approx.)
Average thermal flux	1.49×10^{13} n/cm² sec
Peak thermal flux	3.45×10^{13} n/cm² sec
Steam cycle	Single pressure
Superheater outlet pressure	620–670 psig
Superheater outlet temperature	860° F

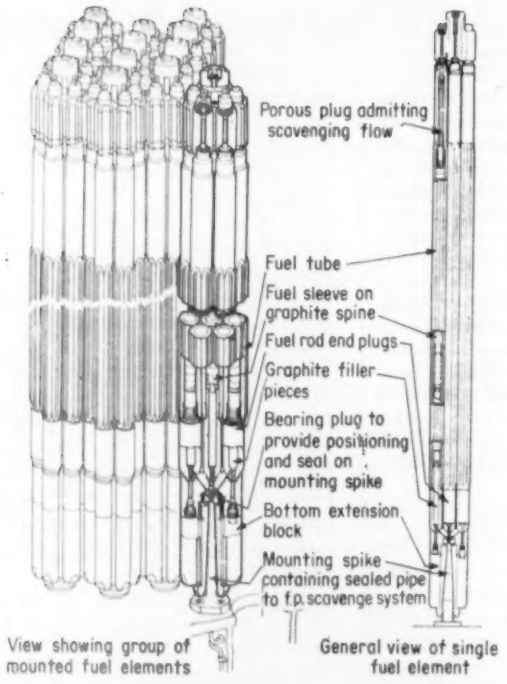

Porous plug admitting scavenging flow
Fuel tube
Fuel sleeve on graphite spine
Fuel rod end plugs
Graphite filler pieces
Bearing plug to provide positioning and seal on mounting spike
Bottom extension block
Mounting spike containing sealed pipe to f.p. scavenge system
View showing group of mounted fuel elements
General view of single fuel element

FIG. 1. Graphite-clad fuel-element assemblies for British high-temperature Th breeder. Gaseous fission products are continuously removed by He bleed flow

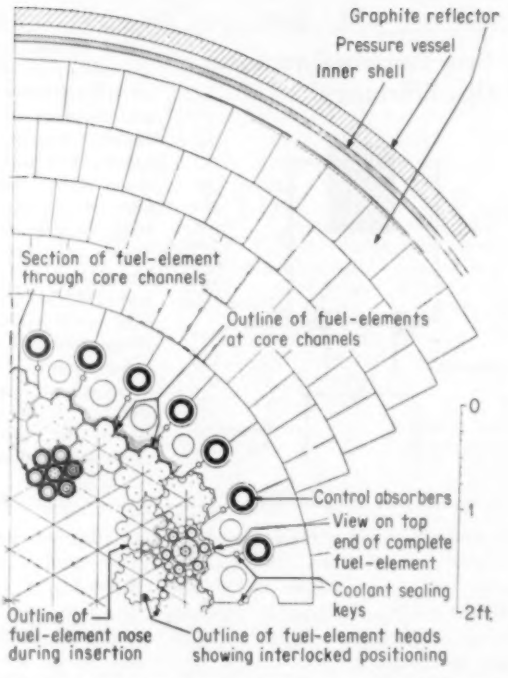


FIG. 2. Core plan showing how seven-rod fuel-element assemblies fit together to define channels for 750° C helium. Graphite serves as both moderator and structure

able conversion or breeding ratio, the fuel must contain highly enriched and fertile materials in the ratio of about 1 atom of U^{235} (or eventually U^{233}) to 20 atoms of Th^{232}. This high fuel concentration suggests a fuel element concept that uses the moderator as a diluent for the fertile and fissionable components. The saving in space makes it easier to increase the total heat transfer surface area so that high power ratings per unit volume of core and per unit mass of fissionable material are made possible. An additional advantage is that fuel elements diluted in this way should be able to accommodate more burnup before suffering serious damage.

As a result of these considerations the AEA has decided to attempt the design of a fuel element system which has no metallic cladding and in which the solid matter of the core consists only of the moderator, fissionable and fertile materials. Although this is an excellent arrangement for neutron conservation, it gives rise to some difficult and unusual problems, the most important being fission product containment.

The choice of moderator material lies between graphite and BeO. Graphite has excellent high temperature thermal and mechanical properties but has poor chemical compatibility with most of the possible coolant gases. BeO on the other hand, while it has inferior thermal properties, is compatible with a very wide range of coolants. BeO is also superior to graphite as a moderator and may be the only one of the two that will give a breeding ratio greater than 1.0.

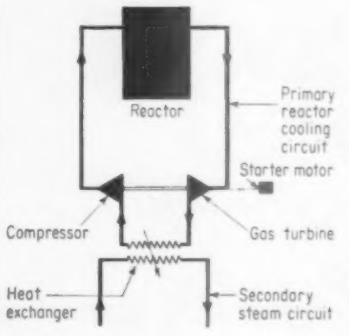
Reactor description. The above ideas have all been pulled together by Harwell in the form of a preliminary design for a 10 Mw reactor experiment. Although work is continuing on both, graphite rather than BeO was chosen as the moderator for the first attempt. This restricts the coolant gas choice to He.

In the core arrangement the concept of a homogeneous mixture of moderator and fuel materials was compromised to the extent of sheathing each element with a layer of inert moderator to act as a fission product barrier. Since the graphite is porous, to prevent fission product escape a reduced pressure is maintained inside the element so that primary coolant flows inward through the sheath, carrying the fission products back into the element. Each element is ducted to a fission product trap to which the inflowing coolant carries the active atoms. The graphite for the sheaths would have to be some 10^5 times less porous than ordinary reactor grade graphite; eventually an impervious seal might be developed.

A possible fuel element design is shown in Fig. 1. Each element is made of seven closely packed rods. In each rod the moderator-fuel mixture forms an annular insert between a graphite sheath and a central graphite spine (alternatively the mixture could be a solid rod within the sheath). A porous plug at the upper end of each fuel rod admits a bleed flow of coolant gas; this is vented at the bottom into a common duct in the block which holds the seven rods.

The core would be made up of 61 seven-rod fuel elements, forming a cylinder roughly 4 ft in length and diameter (see Fig. 2). The closely packed rods have surfaces shaped to define a number of coolant passages which add up to about 13% of the total core volume. Moderator-loaded portions of the elements extend two

Gas-Turbine Circulator Ups Efficiency



The efficiency of gas-cooled reactors may be improved by using a circulator driven by a free running gas turbine rather than the usual electric motor. As shown in the figure, the coolant gas would give up some of its energy to the turbine before reaching the steam generator. Both turbine and compressor may be totally enclosed within the primary circuit. Ideally there is no loss of power in this circuit, as would be the case with an electric-motor-driven compressor. For a reactor outlet temperature of 750° C, the gas temperature at the turbine exit would still be high enough to generate steam at 550° C. The over-all thermal efficiency approaches 40%.

feet past either end of the core sections to form end reflectors. The element assembly is surrounded by a side reflector made up of graphite columns 8 feet in height with an over-all diameter of 10.5 ft.

Both the reflector columns and core elements stand vertically and are pivotally mounted on steel structural members inside a steel pressure-vessel. The helium coolant at 10 atm pressure enters and leaves by ducts near the top of the pressure vessel. The incoming coolant flows downward and maintains the inside of the pressure vessel and the bulk of the side reflector at a temperature not greatly different from the gas at 350° C. It then flows upward through the core and emerges into a Nimonic* chamber at a mixed mean temperature of 750° C. The outflowing gas is transferred through a Nimonic-lined duct to the primary heat exchanger. The Nimonic chambers and ducts are exterally cooled by the incoming coolant and they have no structural function. No part of the primary circuit that has to withstand the full coolant working pressure is normally exposed to gas at more than 350° C.

Since no attempt to generate steam and power is contemplated in this study the secondary cooling water is heated to only about 75° C. This avoids excessive pressurization and allows a small heat exchanger.

* A commercial high-temperature Ni-Cr alloy widely used in the U. K.

The fission product scavenger gas is pumped from the bottom of the fuel-elements through a fission product trapping system using active charcoal absorption beds. The longest period krypton and xenon activities will not be removed. The gas will also go through a chemical purifier to remove O_2, CO_2, CO and N_2O (to prevent graphite oxidation) before returning to the main cooling circuit. The effectiveness of the fission product traps and the chemical purifiers will be the main questions of the reactor experiment. Unless oxidizing impurities in the primary circuit can be kept down to a few parts per million, then carbon mass transfer from the unprotected fuel elements may cause embarrassment. Excessive radioactivity in the primary loop might force impractically stringent requirements on the leak-tightness of the system.

The first step in the hardware phase of this reactor concept will be the construction of a high temperature zero energy reactor at A.E.E. Winfrith to be finished this year. The critical assemblies will be contained in an atmosphere of N_2 which can be heated to very high temperatures with electrical heaters. Core temperatures in the range 800–900° C and reflector temperatures above 400° C will be possible. The facility's initial use will be to carry out tests on the proposed High Temperature Reactor Experiment.

The experimental reactor itself should be ready in about 3 years and commercial stations may be by 1970.

Organic-Cooled D_2O Reactor

The Canadian General Electric Co. (*210*) is studying the merits of an organic-cooled D_2O moderated reactor concept as a possible extension of the NPD-2 concept (a D_2O moderated-and-cooled natural-U reactor). CGE finds that substantial cost savings, relative to an equivalent heavy-water-cooled reactor, would accompany the use of organic coolant. These include:

- No D_2O coolant inventory
- Lower reactor fuel inventory
- No zirconium for core structure
- Low-pressure, low-flow-rate primary coolant system
- Higher thermal efficiency

On the basis of a preliminary design the capital cost of a 150,000 kw(e) plant would be under $300/kw installed. Fuel costs for a natural-UO_2 annular fuel rod with Al cladding would come to about 1.5 mills/kwh at 5,000 Mwd/tonne (4,500 Mwd/ton). The cost for organic coolant makeup is estimated at 0.2 mills/kwh.

In the proposed design the D_2O moderator is contained at 60° C in an aluminum tank pierced by vertical channels or calandria tubes. Aluminum-alloy coolant tubes pass through the vertical channels with a 6.5 mm clearance; this space is filled with CO_2 and serves as insulation between the moderator and coolant systems. Two fuel assemblies stand end-to-end within each coolant tube. The organic coolant, a mixture of terphenyl isomers, enters the core at 575° F and leaves at 800° F. The net station efficiency would be 32%.

Radiation stability of the organic coolant is not seen as a serious problem. For one thing, contrary to the situation in an organic moderated and cooled scheme, most of the radiation energy is absorbed in the D_2O moderator rather than in the organic coolant. Those radiolytic and pyrolytic tars that are produced during operation will be removed in a coolant-purification system by distillation at reduced pressure. A buildup of 20–30% tars in the coolant system is foreseen.

Steam Cycles for Advanced Gas-Cooled Reactors

In line with U. K. plans to elevate gas-cooled operating temperatures (see above), another paper (*273*) considers the probable steam cycle conditions and efficiencies for reactors with gas outlet temperatures in the range 350° C (660° F) to 600° C (1,110° F). They conclude that the steam cycle, originally adopted because it gave reasonable plant efficiencies at the modest coolant temperatures of the first generation reactors, continues to offer attractive efficiencies throughout the range of foreseeable developments.

The table shows the steam cycles and operating conditions that are most appropriate to various stages of development of the gas-cooled reactor. A rapid increase in attainable steam pressure follows upon modest advances in reactor coolant outlet temperature. In turn, the anticipated overall thermal efficiencies show attractive improvements. At the present time, with coolant temperatures limited to about 400° C effective use of feed-heating is not possible since it would be at the expense of steam pressure. As the coolant temperature rises, however, feed-heating can be used to greater advantage and, with foreseeable coolant temperatures of 600° C, cycles working with very high steam pressures and using the maximum amount of feed-heating become feasible.

Much of this potential improvement stems from the fact that, as the reactor coolant outlet temperature is raised, the inlet temperature can be raised as well. How far one can go with the latter depends on the characteristics of the future gas-cooled reactors. The authors make certain conservative assumptions in this connection. One of these is that higher temperature reactors will demand higher coolant pressure drop. Another is that the coolant circulator power consumption will remain 10% of the electrical output of the plant. This is tantamount to conceding that the "nuclear boiler efficiency" will remain at 94%, where it is today.

Higher-temperature, higher efficiency nuclear power plants can, the authors say, pay in two ways which are cumulative: The capital cost of the plant should be less since the thermal output of the reactor will be lower for a given electrical output and the greater fraction of the plant cost lies in the reactor itself. Second, the fuel consumption would be less on account of the higher efficiency.

Advanced Reactor Steam Cycle Data

Coolant outlet temperature	°C	350	400	450	500	550	600
	(°F)	(660)	(750)	(840)	(930)	(1,020)	(1,110)
Coolant inlet temperature*	°C	177	222	260	294	320	343
	(°F)	(350)	(430)	(500)	(560)	(610)	(650)
Type of steam cycle		Dual pressure		Dual pressure with reheat		Super-critical with reheat	
H.P. steam pressure	psi	600	920	2,000	2,800	3,500	5,500
H.P. steam temperature	°C	640	730	820	910	1,000	1,090
	(°F)	(337)	(387)	(438)	(487)	(537)	(590)
L.P. steam pressure	psi	150	230	500	860	—	—
L.P. steam temperature	°F	620	710	820	910	—	—
	(°C)	(327)	(376)	(438)	(487)		
Reheat pressure	psi	—	—	—	—	1,200	1,800
Reheat temperature	°F	—	—	—	—	1,000	1,090
	(°C)					(537)	(590)
Proportion of h.p. steam	%	60	64	67	79	100	100
Feed water temperature	°F	200	375	455	515	520	580
	(°C)	(93)	(191)	(235)	(268)	(272)	(305)
Turbine cycle efficiency	%	28.2	33.5	38.0	41.0	43.5	45.0
Turbine exhaust moisture	%	13.5	14.0	10.5	11.5	11.5	12.0
Station over-all efficiency	%	25	30	34	37	39	40.5
(Efficiency with single-pressure steam cycle; corresponding turbine exhaust moisture)	%	(23.0) (11.0)	(28.0) (12.5)	(31.5) (14.5)	(34.5) †	(36.5) †	(37.5) †

* Coolant temperature at inlet to circulators. † Excessive moisture content.

Pulsed Research Reactor

A research reactor that operates in very short, intense power bursts has been proposed by General Atomic Division of General Dynamics Corp. (*2364*). Like GA's TRIGA, the reactor (which is named FLASH) would be of solid homogeneous type but would use graphite uranium instead of zirconium-uranium hydride.

Under normal operation the reactor would be subcritical and a control rod would be withdrawn to produce a fission burst lasting a few milliseconds with a peak flux and power of about 6×10^{17} n/cm² sec and 80,000 Mw. The power burst is cut off by the extremely prompt negative temperature coefficient which is characteristic of the solid homogeneous core concept. The heat released (800 Mw sec) during the burst is stored adiabatically in the homogeneous core resulting in a temperature rise of 500° C. During a recovery period lasting several minutes up to several hours the heat would be removed by a low-velocity coolant system and the control rod would be repositioned for the next burst. The peak flux when averaged over the subcritical period would give an average flux of the order of 10^{11} n/cm² sec.

GA sees applications for FLASH in:

- Transient tests of reactor components
- Studies of fast-neutron radiation damage in solids
- Radiation effects on temperature-sensitive biological organisms
- Production of short-lived isotopes
- Chemical analysis by neutron activation
- Medical therapy

Advanced Sodium Design from U. S.

Preliminary design and cost estimates were reported by the General Electric Co. (*604*) for a sodium-cooled central-station plant that uses beryllium and graphite-moderator and a novel modular core design. The table lists the pertinent engineering parameters. As shown in Figs. 3*a* and 3*b* the core is made up of thirty-seven subcritical fuel modules dispersed within a graphite moderator and reflector to form a critical mass. Each module is virtually a small tank-type reactor containing both fuel and beryllium moderator and having its own sodium piping connections. The module container is a thin-walled stainless steel tube 13 in. in dia and 26 ft long. Oxide fuel rods form an annulus about a smaller 9-in. dia Be annulus. Cold Na enters the module, flows down past the fuel rods, then up through the central hole in the Be and out the hot piping connection. Sodium level is maintained in each module by a common cover gas system. Modules would be refuelled in place with a specially built machine.

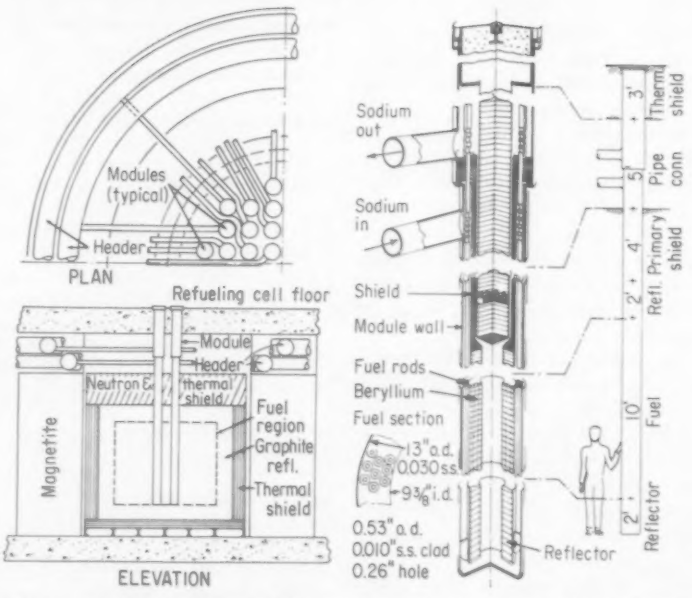


FIG. 3. At left is arrangement of fuel modules that make up core of GE advanced Na reactor. Graphite moderator fills space between individually cooled modules. Detailed composition of typical module is shown at right

When Th is the fertile material, the fuel rod reference design consists of 0.65-in. diameter oxide fuel pellets jacketed in a stainless-steel tube with a 0.010 in. wall. The pellets have a 0.260 in. axial void which provides space for gaseous fission products. The pellets are subdivided into two annular regions; the outer region is 0.0325 in. thick and is highly enriched $U^{235}O_2$ or $U^{233}O_2$ while the inner region is ThO_2. This geometry will sustain a maximum heat flux above 800,000 Btu/hr ft² for a maximum oxide temperature of 4,500° F and a fuel clad surface temperature of 1,100° F. When U^{235} is the fertile component the annular pellets are 0.5 in. in diameter and are uniformly enriched with U^{235}.

Neutron balance calculations show that the $U^{233}O_2$-ThO_2 core, compared with the $U^{235}O_2$-ThO_2 core, has a lower critical mass (625 kg vs. 870 kg) and a higher conversion ratio (0.69 vs. 0.56). Since the conversion ratio is less than 1.0 for both instances, steady-state operation will be a combination of the two cycles unless an independent source of U^{233} is available. The $U^{235}O_2$-$U^{238}O_2$ cycle has a critical mass of 435 kg and an average conversion ratio of 0.54. The U^{233}-Th cycle would have the lowest fuel cost if Pu is sold at $12/gm; the U^{235}-U^{238} cycle would have the lowest cost if Pu can be sold at $30/gm.

The capital charges for the Na plant are estimated 16% higher than those of a light-water boiling reactor plant of the same rating. A large part of this penalty is blamed on the intermediate Na heat exchanger. The U^{235}-U^{238} fuel costs, however, are 25% lower than those of the light water plant, assuming $12/gm for Pu.

Advanced-Sodium-Reactor Design Data

Thermal power	546 Mw(th)
Gross electric power	219 Mw(e)
Net electric power	205 Mw(e)
Reactor inlet temp.	750° F
Reactor outlet temp.	1,000° F
Average burnup	10,000 Mwd/ton
Peak-to-average heat flux	3.0
Number of primary heat exchangers	4
Flow per exchanger	15,100 gpm
Total pressure drop	45 psi
Secondary Na inlet temp.	660° F
Secondary Na outlet temp.	960° F
Steam at turbine throttle	
Pressure	1,450 psig
Temperature	900° F
Flow	1.63×10^6 lb/hr
Be to fuel ratio	2.37
Graphite to fuel ratio	12.9
Secondary flow/loop	11,900 gpm
Total secondary pressure drop	54 psi
Primary piping design pressure	100 psig
Maximum design temperature for piping	1,050° F

INSTRUMENTATION

Low-level Beta Counting

Increased use of H^3 and C^{14} sharpened world interest in low-level β counting. Almost all researchers agreed on both the problem—reducing background to ~1 cpm—and the approach—anticoincidence methods prevailed.

One approach involved eliminating Compton background formed by gammas in the shielding material (*236*). The method: Reduce the amount of material between the counter and the anticoincidence counters. The authors describe three counters: a gas counter for low specific activity, another for low total activity, and a counter for solid samples. All are gas-filled, polystyrene-walled cylinders covered with aluminum. All are concentrically enclosed in stainless steel cylinders, and all use the same counting arrangement—a counting anode ringed by parallel wires which form the anode of an anticoincidence circuit (Fig. 1b). For natural C^{14} counting, the counter was filled with an atmosphere of acetylene, shielded with Hg and paraffin, thus reducing the background to 0.8 cpm, to permit good counting of recent C^{14}, which has an activity of 15 cpm.

Philips researchers emphasized the elimination of cosmic mesons (*2343*). They took a G-M counter and surrounded it with a guard counter (Fig. 1a) so that every meson which strikes the beta counter first has to cross the gas space of the guard counter. The G-M unit is a modified Philips 18505 tube with the cathode reduced to ⅓ the original length to reduce its γ sensitivity, while the beta sensitivity remains constant. The counter is filled with Ne-A-Br_2, which has a rise time considerably shorter than other noble gas halogen mixtures. The background of this counter, too, is below 1 cpm, despite the limitation of the K^{40} background from the mica window.

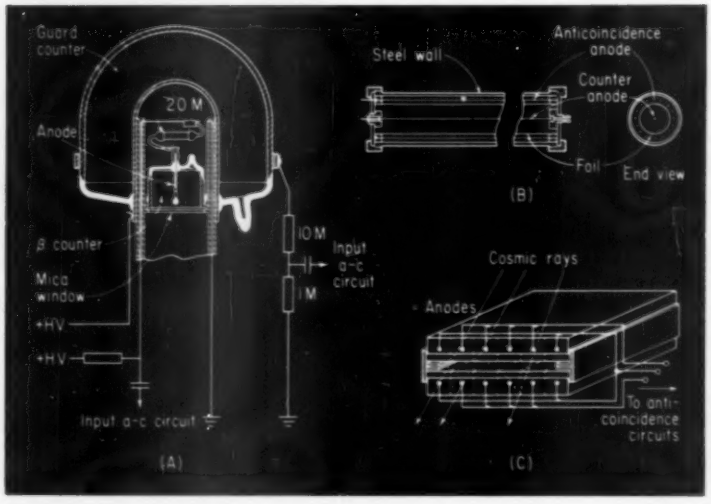


FIG. 1 a. Beta counter surrounded by guard counter
b. Side and front sections of low-activity counter
c. 4π sandwich counter

Another paper from the Netherlands (*554*) describes an essentially 4π sandwich counter made from two flat copper boxes with openings in the top (Fig. 1c). Each opening is covered by a stainless-steel-wire grid, which supports an aluminum window 4.5 mg/cm² thick. The boxes are divided into six compartments by copper partitions to insure that an anode in one compartment will not fire any of the others. The counters are filled with a mixture of 90% argon and 10% alcohol; the total pressure in the chambers is 16 cm Hg. This gives a plateau about 250 v long, with a slope of less than 3% per 100 v. The sample is mounted on thin filter paper so that betas emitted in the downward direction can pass through it and be counted in the lower counter. One other advantage of the setup is that samples such as strontium, calcium, yttrium, or cesium can simply be filtered on thin paper, dried, and counted. The entire counter is shielded by 10 cm of lead, partly taken from an old ship, partly from an old church roof.

The arrangement of the boxes, with the source between them and the windows facing each other, provides a virtually 4π geometry. The counters are wired in anticoincidence to eliminate cosmic ray mesons; the mesons striking only the end chambers are eliminated by sandwich counters which surround the main counter and are wired in anticoincidence with it. The background is thus reduced to roughly 3 cpm, and the β efficiencies are up to 75% for Y^{90}, 55% for Sr^{90}, 73% for Sr^{89}, 55% for Cs^{137} and 40% for a 100-mg Ca^{45} sample with an area of 40 cm².

Gamma Measurement

The γ papers emphasized low-level spectrometry and high-level dosimetry, reflecting the problems of measuring human gamma activity and industrial dosimetry of irradiated foods, plastics, and chemicals. Most of the dosimeters measure total dose rather than dose rate.

One paper discussed the criteria of desirability for dosimeters (*2014*). These include dose range, dose rate, freedom from energy dependence, precision, pre- and post-irradiation stability, independence of temperature effects, and convenience. Although no single system meets all the criteria, four seem outstanding:

- The Fricke dosimeter—Despite its restricted range, it is the most trouble-free and accurate.
- Ceric sulfate—Most nearly meets all dosimetry requirements, but its range is also limited (it must be supplemented below 10^5 rads) and it is not extremely reliable.
- Anthracene—Shows promise for doses over 10^5 rads, but needs further development.
- Glass—Many activated glasses, particularly cobalt F-0621, approach the ideal requirements rather closely, and have the added advantage that they can be fabricated in small sizes. However, long exposures at low dose rates seem to cause fading, so the glasses, too, need further development.

Currently, two dosimeters must be used to span the dose range: Ceric sulfate is best for doses over 10^5 rads, and glass dosimeters are best for lower doses. The Fricke dosimeter would be the choice in the low-dose region if it did not require oxygen-saturated solutions at doses greater than 4×10^4 rads, but it can be used to calibrate other dosimeters in lieu of a primary standard

Viscosity changes and gelation of polymers were also tried in various high-level dosimeters (*2001*). Polyisobutylene, polymethylmethacrylate (Lucite), and copolymers of styrene and isobutylene showed promise, polyisobutylene from 10^4–10^7 rads, Lucite from 10^5–10^7 rads, and styrene copolymers from 5×10^5–10^8 rads.

FIG. 2. Relative positions of crystal and subject found to be independent of the height and weight of the subject

A polyester* dissolved in styrene and sealed in gelatin capsules provides sensitivity from 10^3–10^5 rads (*2009*). The mixture will gel at various doses depending mainly on the relative styrene concentrations.

Antimonate and high-lead glass also show promise for high integrated dosages (*2001*). Several types did not show saturation at 10^9 rads and could possibly be developed for doses from 10^5–10^{10} rads. This range, however, excludes measurement of human γ activity; for this, most investigators rely on γ spectrometry.

With γ spectrometry it is possible to identify, locate, and measure less than 10^{-9} c of γ emitters in the body, and to follow the transport of these elements from one organ to another (*762*). This permits determination, for example, of the clearance rates in the lungs of inhaled radioisotopes.

The spectrometer is a NaI(Tl) crystal coupled to a photomultiplier and a 256-channel analyzer. With the counter in the position shown in Fig. 2, the counting rate is independent of the height and weight of the subject. A collimated crystal permits detection of the distribution of the subject's radioactivity. By noting the rapid changes in counting rate under the photopeaks during scanning, the site of the radioisotope can be determined accurately even though adjacent volumes of the body contain equal quantities of other γ emitters.

The time necessary to measure total body activity can be shortened by using additional crystals. An array of 3 or 4 crystals at appropriate distances in front and behind the subject is very valuable for rapid monitoring of large numbers of people, although one crystal permits measuring an individual's K content with a standard uncertainty less than 5% in a 10-min run. Even if the subject has a large body content of some other radioelement, the data from a 10-min run is sufficient to identify the element present and, in most cases, determine the amount.

* Maleic anhydride, phthalic anhydride and propylene glycol.

Scintillation Counting

Scintillator applications described at Geneva were as varied as the countries they came from: The USA presented a triple-coincidence sorter using an XYZ analyzer; Japan described a radar-type sweeping-collimator survey counter, and Italy showed how plastic scintillators could be used to prepare isodose curves. The Union of South Africa applied liquid scintillation counting to absolute standardization, and the Germans combined scintillography and radiography to project isotope locations directly on anatomical X-ray plates.

The XYZ analyzer was used as a coincidence sorter to display a photographic pattern of coincident radiations on an oscilloscope screen (*647*). The line-and-dot pattern for a given source is unique, and can identify the isotope emitting the radiation, thus permitting rapid, inexpensive analysis (Fig. 3).

The arrangement is very flexible: Set up as in Fig. 4, it can measure γ-γ, β-γ, or triple coincidences, as well as n-capture and backscattered gammas. It permits rapid analysis of fairly complex decay schemes—Eu^{152}, for example—or supplies information about radiative n-capture. A 2-hr run on Hg^{199} and Ti^{49} gave as much infor-

mation about transitions as several weeks of investigation with a 100-channel analyzer.

Isodose curves for multi-field irradiation can be determined directly by using plastic scintillators (*1393*). The fluorescent plastics are photographed and the plates (Fig. 5a) examined with a microphotometer. The gamma curves of the plates permit isodose curves like those shown in Fig. 5b to be made from photometer plate-blackening measurements.

A scintillator and a cathode ray tube provide a survey instrument that spots radiation sources much as radar spots far-off planes (*1358*). The detection unit is a scintillation crystal with a collimator rotating around it at 10 rpm (Fig. 6); the resulting impulses are amplified and displayed on a cathode ray tube. Figure 7 shows both the pattern produced by an isolated point source and the pattern from two such sources close together. If the instrument moves toward or away from the sources, their patterns on the screen move apart or together, respectively. The relative intensity of the source affects peak height, and very strong sources drive the peak off the screen.

A 4π β-γ coincidence counter with a NaI(Tl) scintillator permits absolute standardization of radioactive nuclides (*1104*). In the arrangement shown in Fig. 8, the crystal is mounted as close as possible to the scintillation counter. Glass cells holding 10–12 ml of scintillator are placed in a drop of mineral oil directly on top of the photocathode and covered with a hemispherical mirror. Both β and γ channels are similar electronically; the output from the amplifier's pulse-height selector is fed to a coincidence circuit. The results obtained on Au^{198}, Na^{24}, and I^{131} standards are in excellent agreement with other laboratories and other methods.

Figure 9 shows a scintillation counter arranged to trigger a lamp in response to selected pulses (*975*); the light records the location of radioisotopes and tracers on a previously-exposed X-ray plate. The developed plate shows the exact anatomical location of particular isotope concentrations in the body. The high contrast obtainable through the electronics of the system makes possible the use of smaller tracer doses when *very* small radiation doses are desired, I^{132} may be used successfully.

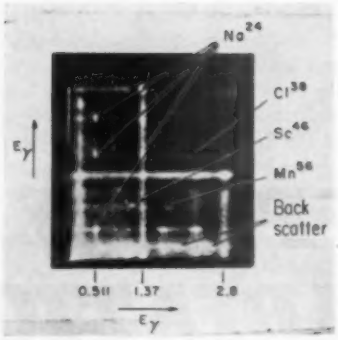


FIG. 3. Gamma-coincidence pattern of n-irradiated tissue showing how characteristic coincidences serve to identify isotopes

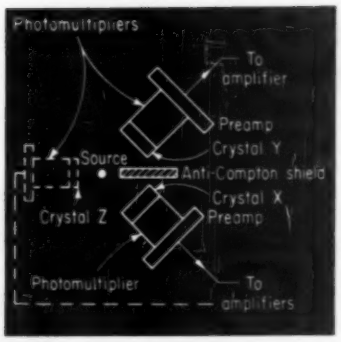


FIG. 4. Crystal arrangement in XYZ analyzer setup. Signals are fed through amplifier-pulse-shaping circuits to show up on oscilloscope

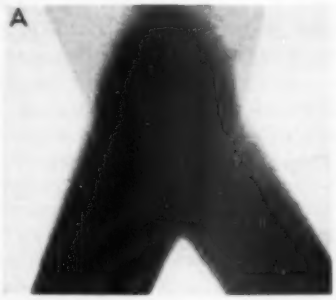


FIG. 5a. Photograph of plastic scintillator irradiated by two sources

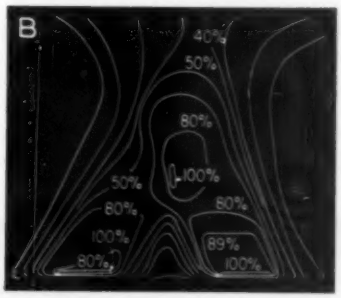


FIG. 5b. Isodose curves made from 5a showing relative doses

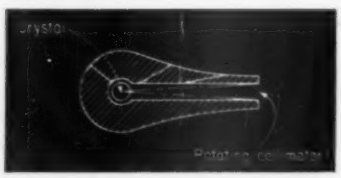


FIG. 6. Scintillation crystal and rotating collimator. Location of source is indicated on cathode-ray tube

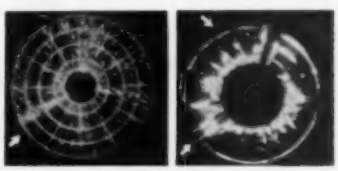


FIG. 7. Pattern of point source (left) and double source (right) with peak (gap) driven off screen

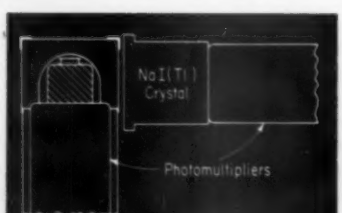


FIG. 8. Arrangement of 4π β counter and scintillation γ counter. The source is placed directly in the liquid scintillator solution

FIG. 9. Combination radiography-scintillography table. X-ray and scintillogram appear on same plate, anatomically in register

Geneva 1958

APPLIED RADIATION

Users of radionuclides and radiation found that what was new at Geneva was largely in new methods for making nuclides and labeled materials and in new knowledge of radiation effects. In contrast, there was not much new in tracing techniques although routine methods have been applied to many more problems of science and engineering.

The large number of papers in applied radiation was surprising. It has been assumed that last September's UNESCO conference devoted to radionuclides in research would reduce the number. But more than 400 of the submitted titles were devoted to such subjects as tracer experiments, radiation effects, new ways to label, detection and dosimetry.

Of all applied-radiation subjects, perhaps high-level radiation is occupying the most attention. Several papers discuss radiation-catalysis processes that appear to be on the verge of economic application. Food irradiation, too, is being actively explored in many countries.

New Labeling Techniques

Advances in labeling techniques are conspicuous among radioisotope papers at Geneva. Subjects include labeled compounds, labeled objects and labeled liquids.

A gas-exposure technique has been developed for producing C^{14}-labeled organics (*284*). Radioactive Kr^{85} supplies the activating radiation; a C^{14}-labeled gas supplies the label; and radical substitution converts an organic substance into labeled organic derivatives.

A few milligrams of labeled carbon dioxide or acetylene, about half a gram of the parent organic, and 2 curies (44 cm^3) of Kr^{85} are placed together in a reaction vessel and held at room temperature for periods of 5–11 days. The table indicates the mixtures used and products obtained.

Two methods are being used for tracing wooden chips in pulp mills (*217, 1101*). In Finland especially made chips are labeled by inserting irradiated copper wires. Canadians soak their chips in a solution of AgI^{131}. AgI soaking is also effective for wood fibers at a later stage in paper making (*216*).

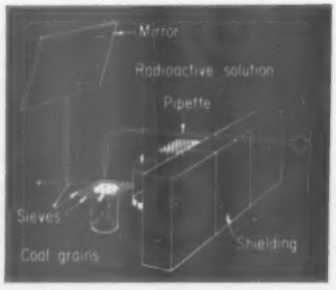


FIG. 1. Coal grains are labeled by soaking in NH_4Br^{82} to study conveyors in West Germany

Br^{82}-labeled coal grains in West Germany have been used to study pneumatic and hydraulic conveyors (*987*). 1-gm, 2-gm and 3-gm grains were weighed and measured and dipped in a solution of ammonium bromide mixed with glue. In another method coal grains were labeled as shown in Fig. 1 with the same labeling solution.

Several experiments have explored surface labeling. Work in Canada has been for ore-processing purposes. In Czechoslovakia surface labeling with inert gases has been found to provide a sensitive method for studying surface

Mixtures and Products in C^{14} Labeling with Kr^{85} (*284*)

Substances in mixture	*Amounts*			*Exposure time (days)*	*Products*
	Organic	*Gas* (*mg*)	(*mc*)		
Benzene and CO_2	0.5 cm^3	2.4	1.3	6	Benzoic acid; 98% of label in carboxyl group. Benzaldehyde labeled in the ring
Pyridine and CO_2	0.5 cm^3	2.3	1.1	5	3 labeled acids; almost all activity in carboxyl group
Pyridine and C_2H_2	0.5 cm^3	0.8	1.4	10	α-, β-, γ-substituted derivatives with C^{14} attached directly to ring
Stearic acid and C_2H_2	0.5 gm	1.1	2.0	11	High-molecular-weight acids soluble in ether

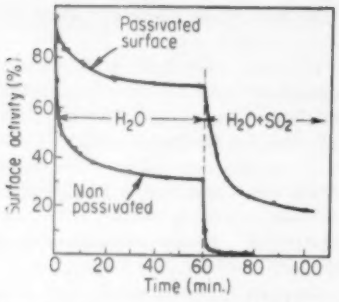


FIG. 2. Escape of inert-gas label from metal surfaces indicates action of corrosive atmosphere

reactions. For surface emission of low-energy beta particles a new method produces an ink-like tritium-titanium suspension.

Swedish experiments. Selective adsorption and subsequent microradioautography were used in Swedish experiments to study metal surfaces (*165*). Several selective absorptions were found to reveal one mineral in the presence of others.

Canadian ore labeling. With metallic ores the Canadian scientists find that some adsorption can always be obtained from any of the metals (*217*). However, they were generally disappointed in their attempts to find labels that would adsorb selectively. They did observe some selective adsorption.

The rare earths turn out to be easiest to use. All metals below zinc in the electromotive series are adsorbed readily.

Inert-gas labels. Xe^{131} and radon labeling through the agency of a high-frequency discharge turns out to be an extraordinarily sensitive indicator of surface reactions (*2444*). It has been used in Czechoslovakia to study the effects of protective or "passivating" films. The degree of adsorption or retention of the label gives a clear indication of the extent of surface reactions long before any corrosion is detectable by other means.

Figure 2 is an example of the escape of the labeling gas and shows the variation that results from exposure of the surface to two corrosive atmospheres.

Tritium paint. The tritium-titanium suspension was developed in Holland (*2347*). Its most remarkable feature is that 90% of the emitted betas escape from the titanium particles in which they are incorporated.

Titanium is vaporized in an argon atmosphere and forms a sooty deposit on the walls of the vessel (Fig. 3). Tritium is admitted and is absorbed in the grains. Finally the grains are dissolved to make an ink-like liquid.

A better label for silt movements appears to be sand irradiated to produce P^{32} (*2357*). Resulting sensitivity was such that experimenters could recognize one radioactive sand grain among 100,000 inactive ones.

The sand was washed in acid, sieved, selected and dried before irradiation. In dispersion experiments it was poured onto the sea floor by skin divers, picked up later on grease-covered cards, and measured with radioautography.

Most determinations of groundwater movements use tritium as the label, either artificially introduced for the purpose or added by nuclear explosions (*176, 1963*). An extensive set of experiments in Israel has evaluated several inorganic salts and labeled organics as ground-water tracers (*1613*).

Laboratory tests with a gravel-filled column compared the following tracers: KCl^{36}, KI^{131}, $(NH_4)_2Ir^{192}Cl_6$ and ethylene diamine tetra-acetate (edta) labeled with Zn^{65}, Ir^{192} and Co^{60}. Trials showed that neither of two Ir^{192} compounds tried was stable enough for the purpose. The compounds were Ir^{192}-edta and $(NH_4)_2IrCl_6$. Good recovery of Co^{60} from $K_3Co(CN)_6$ indicates that $Co^{60}(CN)_6^{---}$ ion is suitable as a water tracer.

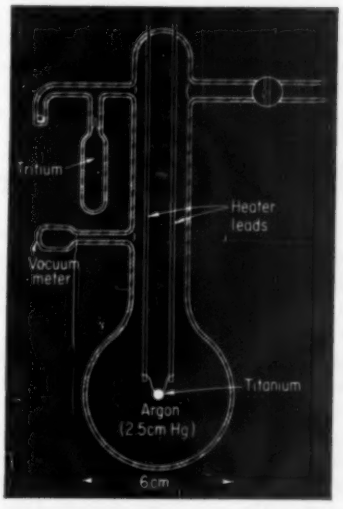


FIG. 3. Tritium-soaked microscopic titanium grains are made with this apparatus for beta-emitting surfaces

Nuclide Production

Many papers from many countries indicate rapid growth in production and separation of nuclides. The greatest amount of work is in fission-product separation reported from the U. S., France and Yugoslavia.

U. S. papers discussed production and use of 1.9-hr F^{18} and a new process for multicurie amounts of the transuranium alpha emitter Cm^{242}. An Indian paper reported the production of high-specific-activity Co^{60} by the Szilard-Chalmers reaction (*1650*).

In the field of stable nuclides, Israeli scientists described an O^{18} plant with an output that exceeds the present world demand. So much B^{10} is on hand in the U. S. that a 500-kg/yr plant is now on standby. Meanwhile in Britain a small B^{10} plant is busily producing B^{10} at a 4-gm/day rate.

An Oak Ridge paper described the first operations of the new multikilocurie pilot plant (*831*). The following long-lived nuclides are already being produced in amounts of hundreds of kilocuries annually: Cs^{137}, Ce^{144}, Pm^{147}, Ru^{106}, Sr^{90}, Tc^{99}. Recovery is from the wastes of two commonly used fuel-recovery processes: Purex and Redox. A test separation of Cs^{137} gave these results: 98.8% recovery of 8,870 curies, specific activity of 22.0 curies/gm, impurities of about 1% each of chlorides of Na, Ru, Fe. Meanwhile a smaller plant is separating short-lived fission products at a rate of 400 curies/month. Its products include Ru^{103}, Zr^{95}-Nb^{95}, Ba^{140}, Sr^{89}, Ce^{141}, Nd^{147}, Pr^{143}, Y^{90}. Oak Ridge is also processing and distributing Kr^{85}, Xe^{133}, Xe^{131m}, and tritium.

An entirely different arrangement is used by the French in a plant for separation of 100-curie batches of Cs^{137} (*1179*). The pilot plant consists of a movable lead-shielded unit that measures 46 × 5 × 6.5 ft. It is separated internally into 11 compartments. External connections are provided.

In anticipation of their own power-

reactor program Yugoslavians have been practicing nuclide separations with mocked-up solutions (*488*). Fission and corrosion products are separated from different concentrations of HCl. Development work has also been directed at isolation of carrier-free I^{131}, P^{32}, S^{35}, Au^{198}, etc. (*467*). Successful procedures have been worked out for S^{35} produced from the reaction $Cl^{35}(n,p)S^{35}$. Contacting with alumina results in 99% adsorption of S^{35} in the bisulfate ion. The final product contains 95% of the original S^{35} and up to 0.1% P^{32} as a phosphate, both carrier-free. The process shows promise for other separations also.

Fluorine chemistry can be studied with the aid of the only usable fluorine tracer, 1.9-hr F^{18}, thanks to methods reported from Oak Ridge (*937*) and Argonne (*1775*). Both groups agree that the best method of production is neutron irradiation of an intimate mixture of LiF and Al_2O_3. The following pair of reactions then produces F^{18}: $Li^6(n,\alpha)T$, $O^{16}(t,n)F^{18}$.

Radiation-processing people are likely to find Cm^{242} a useful radiation source. It emits 6.21-Mev alphas with a 162.5-day half-life and is produced at 100 mg/day by a process reported from the University of California (*1883*).

The process consists of three parts: dissolution of aluminum slugs containing the Cm^{242}, ion exchange to remove impurities, and elution of Cm^{242}. Production is 400 mg of Cm^{242} every four days at a product yield of 85%.

World requirements for two important stable nuclides, O^{18} and B^{10}, appear to be well supplied. As of July 31 the U. S. Atomic Energy Commission has placed in standby a $4-million plant located in Niagara Falls, N. Y., that was designed to produce 500 kg/year of B^{10} at 90–95% enrichment (*1836*). The method used is equilibrium distillation of a BF_3-dimethyl ether complex.

There are four steps: preparation of high-purity dimethyl ether from methanol, formation of liquid dimethyl ether-BF_3 complex, fractionation in a distillation cascade, removal of nonvolatiles.

The small and simple British plant has been producing 90–94% B^{10} for two years (*102*). It uses chemical exchange of the diethyl complex $BF_3(C_2H_5)_2O$.

Oxygen-18. Meanwhile O^{18} is in long supply from Israel (*1612*). 95%-enriched O^{18}, mainly in D_2O^{18}, is produced at rates approaching an upper limit of 11 gm/day by fractional distillation. O^{17} is also produced at a maximum enrichment of 2%—shortly to be increased to 10%.

The British, too, are producing O^{18} with a low-temperature distillation that produces C^{13} simultaneously (*102*). The two are separated into $BaC^{13}O_3$ and H_2O^{18}.

Tracing Technology

In contrast with the variety of new labeling techniques, papers on tracing showed continuing work with tried and true methods. Natural phenomena like ground-water patterns and atmospheric circulation command continuing attention (*176, 403, 404, 1954, 1613, 1614, 1963*).

Dating techniques have been improved by extension of older methods and introduction of dating with Be^{10} (*412*). The ratio of the amount of Be^{10} to the amount of stable Be in a sedimentary deposit tells the age of the deposit.

What makes Be^{10} particularly interesting is its 2.5-million-year half-life. Ages that can be determined with any radionuclide are those in the vicinity of its half-life, and Be^{10} fills an important gap.

C^{14} aging has been improved by a sensitive method for enriching C^{14}-labeled CO, measuring the enrichment of O^{18} in $C^{12}O^{18}$ molecules, and using this to gage the enrichment of C^{14} in $C^{14}O^{16}$ molecules (*1123*).

A large number of papers testifies to increasing use of tracers in biological research. These include a variety of metabolism studies. An exhaustive study of hydrobiology is pointed especially at the concentration of radioactive water contaminants by aquatic organisms (*2317*). Iontophoresis—migration of ions in a biological system under the influence of an electric field—is the subject of a paper from Uruguay (*1528*).

The hydrobiology study, performed in Russia, offers substantial hope that aquatic organisms can be used for decontamination. It is found that most radionuclides pass from reservoir water into the ground and living organisms. The greatest amounts are eventually concentrated in bottom deposits. C^{14}, P^{32}, Ca^{45} and S^{35} have been used in these studies.

Wear studies are continuing and are being especially extended to actions of lubricants and protective coatings (*94*). In Russia the tracers C^{14}, P^{32}, S^{35} and Fe^{59} have been used to study processes taking place on metal surfaces as additives are included in lubricating fluids (*2198*). Special interest has been given to the mechanisms of corrosive wear.

Furnace linings. A new method, introduced in Finland, measures the depletion of blast-furnace linings (*1101*). Each of several holes in the lining is charged with five Co^{60} pellets. It is always possible to tell how many of the pellets remain in any hole and how many have disappeared with lining wear.

Interfacial area. An Oak Ridge study has developed a simple and effective method for studying interfacial area in processes that place immiscible liquids in contact (*832*). It functions by putting an alpha-particle or low-energy-beta emitter in one phase and a particle-sensitive substance in the other. Reaction rate is directly proportional to interfacial area. In the initial development Po^{210} was used as an alpha emitter and detection was by counting neutrons from the $F^{19}(\alpha,n)Na^{22}$ reaction (Fig. 4).

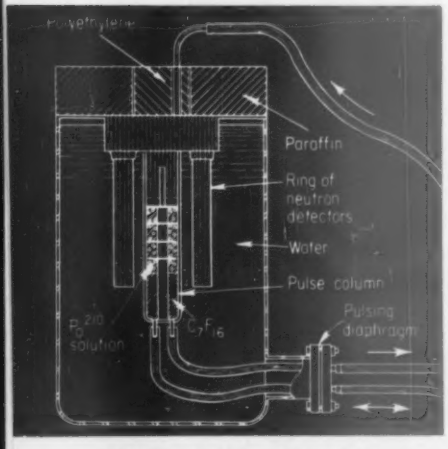


FIG. 4. To measure interfacial areas, put an alpha emitter in one phase, F^{19} in the other, and count neutrons from α,n reactions

Radiation Sources

New types of sources have been described—both large ones for irradiations and small sources for radiography. Probably the most unusual is a reactor designed for irradiation purposes in the U. S. Another interesting reactor irradiator is Armour Research Foundation's fission-gas source. A large irradiator that will use spent fuel from the new EL 3 reactor has been designed in France.

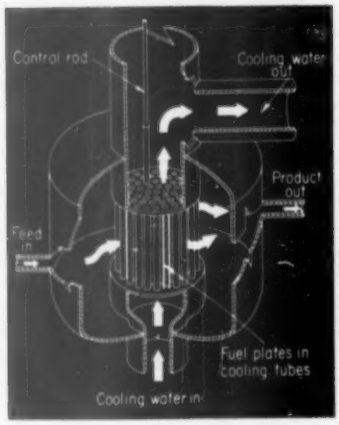


FIG. 5. Irradiation reactor separates process and coolant streams in anticipation of deposits from process stream on reactor surfaces. It is designed to process liquid or gaseous hydrocarbons

Irradiation reactor. Figure 5 shows the reactor designed by Esso engineers in conjunction with Nuclear Development Corp. of America for irradiating a process stream (*795*). This design is based on the assumption that irradiation will produce sediments that will settle out on reactor surfaces. Thus coolant and process streams are isolated from each other.

The reactor is expected to deliver 4.2% of its total power in radiation to the process stream. It will also produce 125-lb/in² process steam. Reactor power level is 70 kw(th).

The Esso people show unusual optimism toward the potentialities of this reactor. They estimate that it can deliver radiation to a liquid process stream at a cost of 2¢/kwh as compared with 79¢/kwh for a Co^{60} source and 44¢/kwh for a linear electron accelerator. If the radiation is delivered to a vapor phase, the cost goes up to 13¢/kwh. Moreover their computations indicate that induced radiation is not great enough to be objectionable in petrochemicals exposed to 1 kwh/barrel ($\sim 0.5 \times 10^6$ rad) and allowed to cool for 30 days.

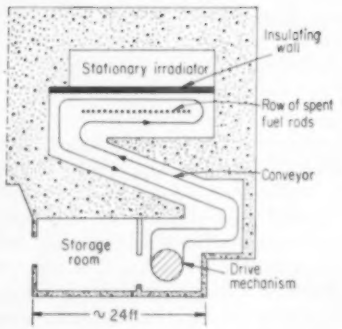


FIG. 6. French spent-fuel irradiator has one stationary facility and another that uses a conveyor belt. Conveyor facility and adjacent storage room are refrigerated

Fission-gas source. Another type of reactor irradiator was described as having significant industrial potential (*1766*). This is the fission-gas chamber operated with the Armour aqueous homogeneous research reactor. A flushing stream of oxygen removes fission gases plus hydrogen and oxygen liberated by radiolysis. The radiation source is a 12-in.-diameter cylindrical tank holding 75% of the reactor gas volume. With reactor operation at 10 kw the source builds up to ~1,200 curies and produces a radiation intensity of 10,000 r/hr. The associated neutron field, mostly from core leakage, is 10^4–10^5 n/cm²/sec. Effective gamma photon energy is 1.3 Mev.

Spent-fuel irradiator. The French design for a spent-fuel irradiator is shown in Fig. 6 (*1212*). It combines two facilities. In one a stationary array of nine rows of rods and exposure spaces will provide dose rates of 95,000–800,000 r/hr.

The other facility has a conveyor that carries packaged samples past a row of 20 rods, turning them once in transit. The rods are suspended in a cold room kept at −7° C, and adjacent to this is a storage room at 0° C. The carrier is designed for packages that have a 16 × 27.5-in. cross section.

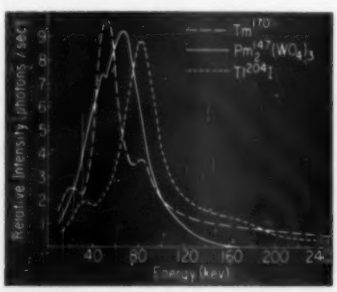


FIG. 7. Comparison of spectra shows $Pm^{147}{}_2(WO_4)_3$ is superior to Tm^{170} and $Tl^{204}I$ because of absence of high-energy photons

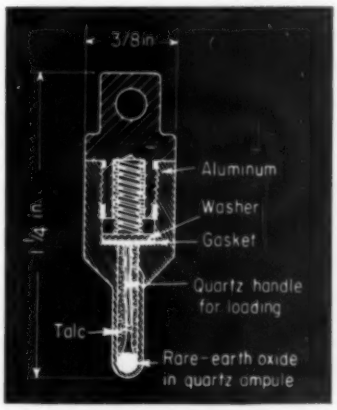


FIG. 8. Radiography source has activated rare-earth oxide or Tm^{170} in quartz ampule, contained, in turn, in an aluminum tube

Radiography sources. A variety of radionuclides and methods of preparation are recommended for radiography sources. One Russian paper described the use of Tm^{170}, Eu^{155} and Ce^{144} for thin-wall radiography (*2235*). On the other hand, a U. S. group considered Tm^{170} usable for diagnostic radiography but brehmsstrahlung sources much better (*828*). They found that the high-energy brehmsstrahlung generated by Tm^{170} reduces the available contrast. They made better sources by mixing Pm^{147} and Tl^{204} with materials that function as brehmsstrahlung generators. Since intimate contact between beta emitter and brehmsstrahlung target is advantageous, they tried the chemical compound $Pm^{147}{}_2(WO_4)_3$ and found that its spectrum was free of the damaging high-energy components of other sources. Their spectra are shown in Fig. 7.

Another U. S. paper reported a special small-source design that is recommended for radiography with small source-film separation (*829*). A

special quartz ampule (Fig. 8) was filled with a target nuclide—either Sm_2O_3 or Gd_2O_3—and irradiated in Canada's NRX to 10^{19} n/cm². The 8.5-mg loadings contained 81.5% Sm^{144} and 15% of Gd^{155}, respectively, and irradiation produced Sm^{145} and Gd^{153}. A similar source was loaded with Tm^{170}.

Radiation Effects

An atmosphere of activity and anticipation is characteristic of worldwide studies in effects of radiation. Damage mechanisms were the subject of many papers, and it is conspicuous that the field is still an open one. New questions are being formulated as fast as answers are being found. Several papers reported significant studies of radiation-induced chemical reactions. Several were concerned with means of protecting against radiation damage in both living and nonliving matter.

Mechanisms. Two American papers reported careful and revealing work on radiation effects in water. Decomposition has been found to be highly dependent on rate of bubbling (*952*). The *G* value for hydrogen production is found to be 0.4 molecules/100 ev with agitated bubbling and less than 3% of this with no bubbling. The explanation is that bubbling permits escape of the hydrogen and prevents its combination with water to form H_2O_2. The rate is not appreciably affected by the presence of NaCl, KBr and KI, but $G(O_2)$ is greatly decreased by the presence of the halides.

Alpha-particle irradiation of triply distilled water containing only a little CO_2 gave the following *G* values: $G(O_2) = 0.10$, $G(H_2) = 1.45$, $G(H_2O_2) = 1.28$ (*1128*). An observed reduction of H_2O_2 yield with I^- concentration agrees with predictions of a one-radical model and cylindrical diffusion of the ions. The indication is that an electron is transferred in each encounter of a neutral OH radical and I^-.

A Russian paper showed good agreement between theoretical predictions concerning crosslinking and observed effects (*2294*). Another paper on effects in organics offered a comprehensive review of much radiation chemistry (*1516*). It described investigations of effects of radical scavengers and self-scavenging in solutions that demonstrate changing yields with changing dose rate. A U. S. paper described extensive studies of the radiation chemistry of organics and halogenated organics with the use of deuterium and radioactive halides (*928*).

The introduction of new magnetic-resonance techniques to study radiation-damage mechanism was exemplified by one paper from the U. S. (*948*). The author uses electron spin resonance to measure free-radical populations. The results with *n*-alkanes and simple alcohols correspond with radiation-chemistry information, but with some compounds they are in contradiction with previous work. Particularly in plastics it is thought that the contradiction may result from unknown differences in starting materials.

A West German paper offered an exhaustive essay on damage mechanisms with specific reference to extensive studies on copper (*998*). The author suggests three stages for damage studies: calculations of the number of defects, theoretical treatment of the effect of defects, and theoretical treatment of annealing of defects. An American paper also dealt with irradiation of copper (*2385*).

Induced reactions. Profitable irradiations that promote useful chemical reactions are a near thing if one can take seriously the implicit and explicit suggestions of a number of papers. One of the most exciting is a U. S. study of sulfoxidation of hydrocarbons to produce sulfonic acids (*797*). *G* values as high as 6,500 have been demonstrated. Normal paraffins from hexane to hexadecane can be sulfonized with good yield. Long-chain molecules with up to 1,000 carbon links have been produced.

These investigators have made some interesting discoveries about inhibition of radiation effects. Several hydrocarbons that are particularly unreactive in sulfoxidation reactions are effective inhibitors if they are mixed with reactive ones. For example, 10% of 2,3-dimethylbutane reduces the *G* value for *n*-hexane from 4,880 to 53.5. The inhibition mechanism is the formation of H_2O_2.

Another U. S. paper reported significant increase in reaction rate between propane and oxygen (*794*). 800 curies of Co^{60} delivered 280,000 r/hr to the gas mixture, which had a residence time of 2, 5 or 10 sec in the irradiator. Yields were increased by 100–200% over nonirradiated reactions, apparently because of increased free-radical concentrations.

Effectiveness of iron catalysts has been increased by a factor of about 40% by irradiation to $\sim 4 \times 10^7$ rads (*63*). No good explanation is available, but a pertinent observation is increased emission of low-energy photoelectrons. Another paper that mentioned similar effects on electron behavior reported the effects of intense pulses from a 7-Mev linear accelerator on circuit components (*802*).

A British team described extensive studies devoted to nitrogen fixation (NU, July '56, p. 22) (*76*). They have found a value of 0.9 for $G(NO_2^- + NO_3^-)$ and suggest that it can be increased to 5.0 with proper gas mixtures and conditions. For economic production a *G* of 10 should be achieved. The reactor they suggest for the purpose would have its fuel in a gas-fluidized slurry. High decontamination (10^7–10^8) would be required, but it can be achieved with distillation processes.

Radiation protection. The subject of radiation protection was often discussed. Two of the papers already mentioned (*797*, *1516*) included significant discoveries on this subject. One paper from Germany indicated that experiments with hibernating dormice had confirmed radiation protection by cysteine (*993*). Another German paper indicated that sulfhydryl protective agents operated not by reducing radiation effects but by stimulating regeneration in resting cells (*991*).

Radiation resistance in wool is important to people who would like to sterilize wool products with radiation. Sterilizing doses have been found to cause some deterioration. But wool treated with 1,5-difluoro-2,4-dinitrobenzene or quinone is protected against deterioration (*972*). In untreated wool 10^7 r caused a 12% decrease in tensile strength. No change was recognized in the treated material. An American paper reported neutron and gamma radiation had similar effects in cotton, wool, silk, and synthetic fibers (*821*). Wool is relatively resistant, but this paper, too, suggests that treatment might improve its resistance.

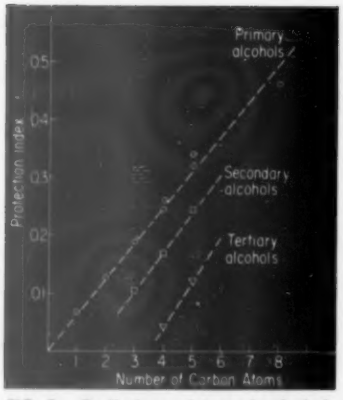


FIG. 9. Radiation resistance in alcohols is a function of both the number of carbon atoms and the type of substitution in the molecule

A study of the influence of molecular structure on radiation resistance found some significant correlations (*915*). As indicated by Fig. 9, protection indices in alcohols with the same number of carbons decrease in the following order: primary, secondary, tertiary. In any one type of alcohol, indices increase linearly with the number of carbons.

Oils and greases, too, can be made radiation resistant (*2384*). Alkyl aromatics make the best base materials, and additives such as selenium compounds offer protection against radiolysis. Resulting oils withstand 10^{10} r as compared with the usual solidification at 5×10^8–5×10^9 r.

Plastics. Papers on irradiation of plastics, rubbers and elastomers came from Russia, Germany, Japan, the United Kingdom and the U. S. The Russians have been studying vulcanization with 80-kv X-rays and a 21,000-curie Co^{60} irradiator (*2085*). They are presently building a 16,000-curie Co^{60} unit especially for this work. A U. S. paper indicates that the dose required for vulcanization can be significantly reduced with additives. Incorporation of 15–20% of 2,5-dichlorostyrene reduces the dose requirement from 33.4 to 1.1 Megarad (*826*).

Several papers discuss copolymerization (*65, 826, 1346, 1350*). The most practical application mentioned appears to be the grafting of material to the synthetic fiber "Vinylone" in Japan to increase dye retention.

Food Preservation

Food-irradiation papers at Geneva indicated a difference of emphasis between the U. S. program and those of other countries. The U. S. appears to be mainly concerned with long-range, large-scale programs and sterilizing doses. Other countries are spending time on smaller programs, smaller doses, short-term preservation and immediate application.

Two papers indicate present-day usefulness. Russia has already cleared irradiated potatoes for sale (see box on this page). British experiments indicate that irradiation can solve a current problem: food poisoning and paratyphoid from imported frozen eggs. Several papers offer confirmation that radioactivity will offer no problem.

Vegetables. Work from several countries is in substantial agreement on preservation of vegetables, particularly potatoes. Minor disagreements are evident concerning optimum dose levels. The French recommend 7,500 rads for sprout inhibitions (*1602*). The Russians are using 10,000 rads (*2131*), and Norwegians recommend as high as 15,000 r (*568*). The French object that doses of 10,000 rads and above enhance deterioration, and the Russians offer some confirmation of this observation. A Swiss paper reported work on combined effects of radiation and antibiotics (*258*). The British have been investigating this for fish (*66*).

The Norwegians found beneficial effects from 5,000–10,000-r doses delivered to carrots, apples, and bananas. However, while 5,000 and 10,000 r doses decreased rotting in carrots, 15,000 r markedly increased it. The explanation is that the higher dose apparently damages the outer cells that are essential to rot resistance.

Frozen eggs. The most promising and compelling use for food irradiation appears to be the treatment of frozen eggs that are imported in large quantities in Britain for use in bakeries, confectioneries, etc. Salmonellas that lead to food poisoning and paratyphoid appear rarely in eggs, but one infected egg appears to be capable of badly infecting a quantity of dried and frozen yolks or whites. Tins of frozen egg materials have been exposed to gamma doses of 300,000–500,000 r (*64*). No salmonellas remain after irradiation.

Radioactivity. One of several U. S. food-irradiation papers described investigations of induced radioactivity in foods treated with 25-Mev electrons (*807*). Short-lived C^{11} and Cl^{34} offer an immediate hazard and require remote handling. After a 5-hr cooling, however, no hazard remains even at levels significant for ingestion.

The Norwegians reached the same conclusions in their irradiations, which were performed in their JEEP reactor (*568*). Their fields consisted mostly of photons, but some neutrons were inevitable. They found no serious radioactivity following 15,000-r exposures, and no measurable activity a few hours after exposure.

Russians First to Permit Sale of Irradiated Food

A Russian "first" has been announced at Geneva. In a paper that was not presented orally but will appear in the proceedings, they make the following statement (*2131*): "On the basis of the tests conducted, the USSR Chief of Public Health Inspection has authorized the use of potatoes irradiated with a 10,000 r dose for food." The tests referred to were exhaustive investigations of radiation effects on chemistry, odor, taste and cooking quality of the potatoes.

No irradiated food has been cleared for general consumption in this country, although current tests show that several are likely to be found acceptable. It has been conjectured that the first food cleared by the U. S. Food and Drug Administration is likely to be a sterilized food rather than one subjected to a low dose.

The Russian paper indicates that their potato work has been going on since 1955 and makes reference to American work on the subject published in 1954 by A. H. Sparrow and E. Christensen (NU, August '54, p. 16).

Geneva 1958

SHIP PROPULSION

In nuclear propulsion for merchant ships—as in central-station nuclear power—the three years since 1955 have seen the transition from the dreams-and-plans stage to the first tangible realizations. Russia and the U. S., who still hold a lead of several years on the rest of the world, reported on the first and second non-military nuclear surface ships: the icebreaker *Lenin* and the passenger-cargo ship *Savannah*. Britain, France and Japan gave evidence of their high interest in this field with some economic studies and original designs. On the other hand Germany, Holland, Norway, Sweden and Denmark, who have also been active in at least studying nuclear ship propulsion (NU, Nov. '57, 86), made no contribution at Geneva in this field.

Lenin, Savannah

The Soviet Union (*2140*) let out the secret, well kept for three years, that the 44,000 horsepower main turbines of the 16,000-ton *Lenin* are fed, not by one, but by three reactors. These are pressurized water units, not graphite-moderated water-cooled as is the first 5-Mw(e) Russian atomic central station. Each reactor has an independent primary circuit carrying coolant water to a separate steam generator, where superheated steam is made in the secondary, water loop; this is fed to the four main turbogenerators (11,000 hp each), to auxiliary turbogenerators and auxiliary steam mechanisms. The turbogenerators produce direct current for electromotors that rotate the three propeller shafts; the auxiliary turbines produce a-c of commercial frequency. Shielding is of layers of iron and water, with concrete replacing the latter where installation of water tanks is impractical. The primary loop area is kept at low pressure by exhaust fans to prevent spread of radioactive materials into working areas: the air is discharged through the hollow mainmast. If necessary (as when doing maintenance) active water may be discharged from the primary loop into shielded tanks; thence it is dumped into the sea only after cooling and filtering in ion-exchange columns.

Lenin can steam a year without refueling; she will have a cruising speed of 18 knots in open water, 2 knots through 6-ft-thick ice fields.

Savannah's characteristics (*1023*) have of course been described before (NU, Nov. '57, 79); her builders go on to discuss basic design decisions on:

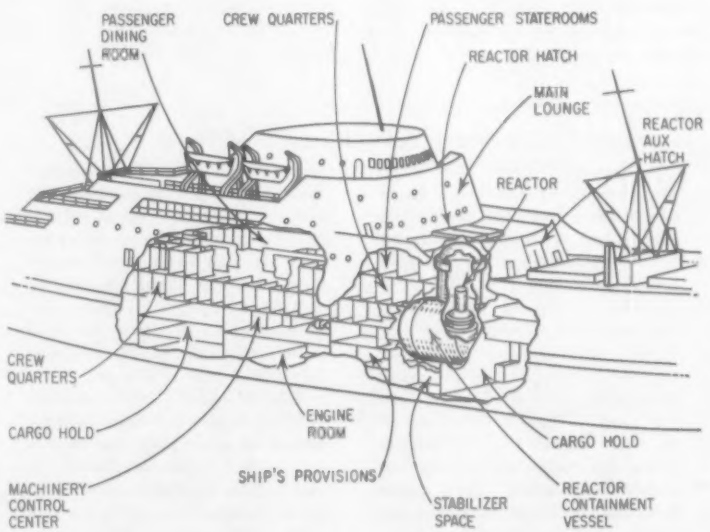
Shielding—The primary shield of 33 in. light water and 3 in. lead around the pressure vessel reduces radiation to levels comparable with that from unshielded primary piping; 30 min after shutdown the radiation is 200 mr. The secondary shield is comprised of 6½ in. lead plus 6½ in. polyethylene above the containment vessel, vertical walls of 4-ft concrete enclosing its lower portion, and salt-water ballast and fresh-water tanks 5 ft deep below it.

Containment—The 35-ft dia, 51-ft long containment vessel of 2½-in. steel, hermetically sealed and designed to withstand 186 psia, will contain all primary components and auxiliaries. It can resist the maximum pressure if all primary water (77,000 lb) were flashed to steam.

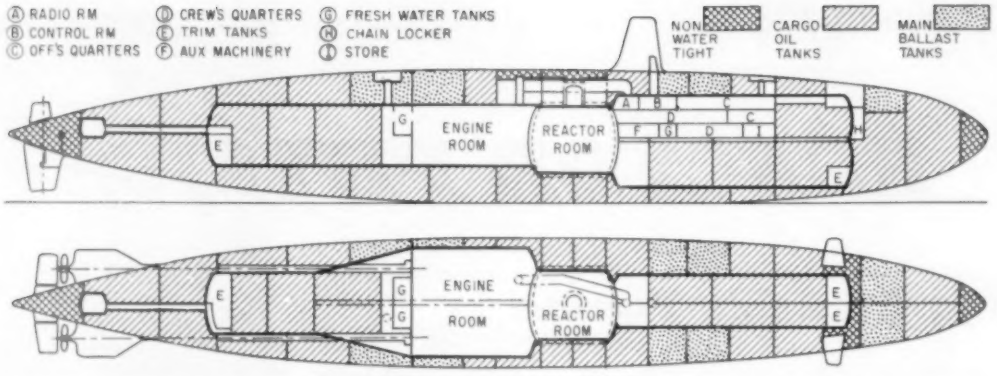
Hazards—If the primary were ruptured and cooling lost, and core meltdown resulted, then—even if all fuel elements melted simultaneously, an unlikely assumption—the melted core could burn through the bottom of the reactor vessel in 2½–3½ hr after loss of cooling, but would be safely held within the containment vessel.

Other Projects

Japan submitted two interesting specific designs. One for a nuclear submarine tanker (*1320; see sketch, right*) is from Mitsubishi's Kobe Shipyard. Before this design can be built, some major problems remain to be solved, the paper states, including: per-



N. S. SAVANNAH cutaway showing relationship of reactor to ship's spaces



JAPAN'S SUBMARINE TANKER DESIGN shown in elevation and plan. The 590-ft, 48,200-displacement-ton tanker can carry 30,000 dwt tons of cargo oil at 22 knots submerged cruising speed. It would be powered by a pressurized-water reactor of 180 Mw(th): loading is 7,270 kg U as UO_2, enriched to 1.7%, pelletized, zircaloy-clad and assembled in 88 elements; 17 cruciform cadmium-silver rods provide control. Refueling is once in two years (after 10,000 hr full-load operation); burnup is 9,200 Mwd/ton. Primary coolant inlet temperature is 538° F, outlet 497°. Two vertical shell-and-tube steam generators feed 566 psi, 482° F steam to two cross-compound turbines each putting out 22,000 shaft horsepower to its screw. Complement is 13 officers, 37 crew. The craft normally navigates underwater (it can safely operate at 328 ft depth) but can also operate at periscope depth, or surfaced when entering port, loading and unloading, and traversing lanes of heavy sea-traffic. Heavy lines on diagram indicate pressure-resisting hull; most of cargo oil tanks are outside it in non-pressure-resistant hull

fection of nautical instruments for underwater navigation; "counter-radioactivity measures in the completely-enclosed hull; reactor plant cooling system during underwater navigation."

The other (*1319*) is for a 20,100-ton, 23.5-knot emigrant ship. Because of population pressure, Japanese have been emigrating to Latin America for 50 years, but there are now only five ships equipped for this service: their capacity of 8,000 emigrants/yr needs to be quadrupled, their transit time of 40 days one-way reduced. Hence the design for a nuclear ship to carry 2,300 emigrants, 200 passengers and 8,000 tons cargo outbound, 400 passengers and cargo back to Japan. She would have one 180-Mw(e) pressurized-water reactor, having the same core and coolant characteristics as that for the submarine tanker. Shielding and containment principles by and large follow *Savannah*.

The ship's earning power would make it a profitable venture. It could make five round-the-world trips a year from Kobe to the east coast of South America and back via San Francisco, saving 2,900 sea miles, 31 days at sea and 22 days in port per voyage as compared with a conventional ship, which can make only three trips/yr. Capital cost of the ships is placed at $24-million, depreciable over 20 years; fuel cost is set at $1.5-million/yr; wages for a crew of 232, supplies, insurance, repair and miscellaneous expenses would run about $1.2-million. Passenger and freight income is set at $6,470,000/yr, direct charges for passengers and freight at $1,442,000/yr, leaving earnings of $5,028,000/yr. Because of Japan's capital cost structure the ship would make no profit until its 7th year, but in its 13th year and thereafter (when loans are repaid and no interest due) the profit ratio "will well exceed 40%"; aggregate total profit over 20 years comes to $18.6-million and average rate of profit on invested capital is 17%. The smaller, slower *Argentina Maru*, most modern emigrant ship in service, earns profits from her first year; these add up over 20 years to $9.7-million at 34% average profit rate. "It may be concluded from the above that a nuclear-powered emigrant vessel does not compare favorably in profit ratio with *Argentina Maru*," say the authors, "yet it may no less confidently be concluded that a nuclear propelled emigrant vessel is after all a sound proposition."

Considering in more abstract terms the shipping field at large rather than a specific trade-route, the British Shipbuilding Research Assn. (*265*) reaches a far different conclusion. "For the oil tanker there appears to be no special advantage to be gained from nuclear propulsion . . . We can assume that the days on voyage and days in port are the same [for a nuclear and conventional ship] . . . It appears that the capital cost of a pressurized water reactor is too high and the thermal efficiency too low for this system to be attractive for propulsion of commercial ships, using slightly-enriched fuel. The gas-cooled systems should give better thermal efficiency but it is by no means certain that the size and cost of those systems can be reduced to acceptable values without the enrichment and hence the fuel cost becoming excessive . . . Immediate prospects for commercial nuclear-powered ships do not appear to be very bright . . ." The report does raise the possibility of eventual use of plutonium as a marine nuclear fuel. Both it and another British paper (*266*) by the U. K. Atomic Energy Authority agree that economically-attractive marine nuclear power plants of less than 20–30,000 shp are unlikely.

Results of a two-year study in France by industrial groups for the French AEC and merchant marine on the application of three reactor types to shipboard use (pressurized and boiling water by France-Atome, gas-cooled by Indatom) are reported (*1137*). Common ground rules set for the studies specified a 40,000 displacement-ton single-screw tanker with 20,000 hp normal maximum. No conclusion is offered as to relative merits of the types.

REACTOR PHYSICS

Using Burnable Poisons

A summary (*1900*) of work at Westinghouse and Knolls Atomic Power Laboratory concludes that combinations of self-shielded burnable posions can match the reactivity requirements over core life for a given fuel loading and at the same time avoid an excessive residue of poison at end of life (which costs in fuel inventory). This allows a considerable extension of core life for a given amount of shim control. In addition burnable poisons can be used to flatten power distribution peaks.

To date boron has proved to be the most useful burnable poison. It has a microscopic cross section sufficiently large to leave little residue even for moderate fuel burnup, and yet requires only a small amount of shim control. While the He gas from the B^{10} disintegration causes metallurgical problems, other materials such as Hg or Li which have suitable cross sections have even greater engineering problems.

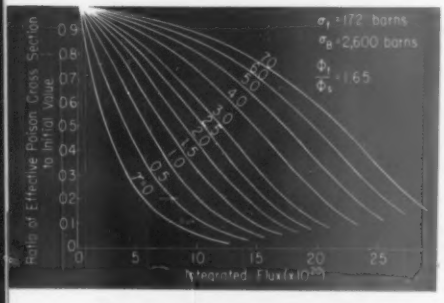


FIG. 1. Effective cross section vs. burnup for self-shielded boron

Burnable Poison Concept

Ordinarily the amount of excess fuel that can be loaded into a reactor is limited by the available shim control. If the core is loaded with a neutron poison, however, the reactivity worth of a gram of fuel is decreased so that the initial fuel charge can be larger for the same amount of shim control. If the burnup of poison is appreciable during core life, both the absolute and fractional fuel burnup will be greater than for the unpoisoned core. The price paid is the extra fuel needed at end of life to compensate the unburned poison (some poison will always be left).

The poisons of greatest practical interest are those with microscopic cross sections greater than the fuel cross section. Such poisons will burn out at a greater rate than the fuel, giving a minimum residue at end of life. When the concentration of a poison of this kind exceeds a certain level, the poison burnout is so great that the core reactivity will first begin to *increase;* later, when burnout has reduced the reactivity effect of the poison, the effect of fuel burnup will cause it to fall again in the usual way. This suggests an alternate mode of shim-control operation that will lead to greater core life. Enough of the high cross section poison is loaded so that at startup the shim rods are completely withdrawn. As burnup proceeds, the shims are first inserted to compensate for the positive reactivity transient and then later withdrawn as reactivity falls. In this way much larger core loadings can be accommodated for the same amount of shim control. The limit on fuel loading for a given poison cross section is set by the amount of positive transient the shim control can compensate for. More fuel loading can be had with a smaller poison cross section (but still larger than the fuel cross section) at the expense of a larger end-of-life residue.

Poison Self-Shielding

In discussing the optimum value for the burnable poison microscopic cross section, we have tacitly assumed that poison elements could be found with any desired cross section. Actually this is far from the truth—nature has provided very few elements with both suitable engineering properties and absorption cross sections that allow burnout at desirable rates. Fortunately, however, the effective burnout cross sections of available poisons can be altered to some extent by using the poison in the form of lumps or other geometrical shapes that give a significant amount of self shielding. With high cross-section poisons, such as boron for instance, we have seen that the obtainable core life is finally limited because the boron burns out too fast (causing the reactivity to go too high for fuel loads above a certain value). By self shielding the boron, its burnout rate can be reduced, thus allowing further extension of core life.

The self shielding factor g (defined as the ratio of the effective cross section to the unshielded cross section) is given approximately by

$$g = (1 + \gamma N/N_0)^{-1}$$

where N and N_0 are the concentration of atoms in the poison lump at time t and at beginning of life respectively. The constant γ is a function of the lump geometry, N_0 and σ_p. For a sphere immersed in an infinite homogeneous medium $\gamma = \frac{3}{8}(2rN_0\gamma_p)$, where r is the sphere radius. In actual practice γ must be calculated for the particular core configuration and thermal spectrum.

Figure 1 shows the time dependent effective cross section for boron over a range of values of γ. As γ increases the poison cross section tends to fall more slowly at the beginning of burnup but much more rapidly near the end of burnup. The γ shielding parameters in Fig. 1 include the effect of epithermal absorption. (The values assumed for the effective epithermal cross section $\sigma_I = 172$ barns and the fast-to-slow

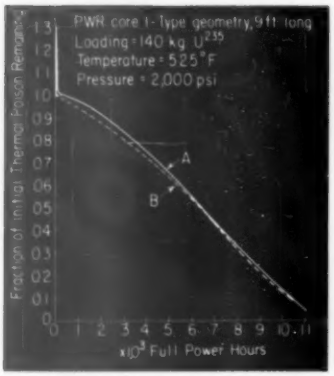


FIG. 2. Lifetime comparison of required shim control (A) and poison control (B) for B^{10}-loaded PWR core

flux ratio $\phi_t/\phi_s = 1.65$ are typical of small highly enriched light-water cores.) The epithermal absorption causes a marked reduction in the effective shielding of the poison and is an important factor in practical design. (With only thermal absorption the curves in Fig. 1 would have γ values of roughly ⅔ those shown.)

With a high cross-section unshielded poison, we have seen that as the poison burns up the reactivity first rises and then falls. If the same poison is heavily self shielded, its effectiveness changes very little with burnout at first so that the reactivity will *fall* in the beginning. Later, as the poison becomes depleted, its self-shielding decreases and the poison burns out faster than the fuel, causing a reactivity rise. Still later the reactivity will fall as with the unshielded poison.

One would like to adjust γ so the initial dip and subsequent rise in reactivity are equal and thus obtain the minimum reactivity mismatch. Nomographs have been developed at KAPL that give values of fuel loading, boron loading and γ that satisfy this condition. Even smaller mismatch can be had by using a combination of a lightly

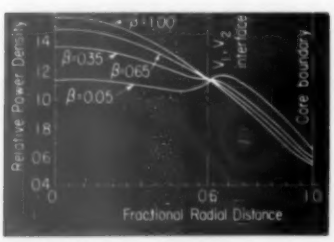


FIG. 3. Flux flattening with two-region loading of burnable poison

shielded and a heavily shielded element. The initial positive-reactivity increase from the lightly shielded poison will tend to cancel out the initial reactivity dip of the heavily shielded component. Figure 2 shows the excellent match obtained for a PWR core using B^{10} poison, 95% of which had $\gamma = 2.5$ and 5% with $\gamma = 0.5$. This calculation took into account fast neutron depletion of the poison and the variation of the fast to slow flux ratio.

Spatial Distribution

So far we have considered the poison uniformly distributed in the core. However, after selection of the optimum poison cross section and concentration, core life can be extended still further by distributing the poison so as to counteract nonuniform burnup.

If we consider two cores of equal fuel loading, one having burnable poisons and one not, the core without the burnable poisons will usually have a greater nonuniformity of fuel burnup since the control rods will have to be inserted much farther in the core during most of the core life and will introduce greater flux distortions. Nonuniformity of fuel burnup usually results in loss of reactivity—the fuel tends to burn out most in regions of high thermal flux which often are also regions of high statistical weight. In a uniformly loaded burnable poison core the poison will burn out most rapidly where the fuel does, thus tending to partly offset the reactivity loss due to the nonuniformity of fuel burnup. This loss is even further reduced by distributing the poison so that it has greater concentration in high burnup regions.

This principle is illustrated by calculations done at KAPL on a cylindrical light-water moderated and cooled reactor with a highly enriched zirconium core. The reactor was shimmed by inserting the control rod bank in the top of the core. Since the flux is low at this end of the core, it makes sense not to put any burnable poison in this region. Calculations for (a) uniform distribution of boron poison in the core and for (b) the same amount of poison distributed in the lower ¾ of the core showed an improvement of 20% in maximum-to-minimum fuel burnout for scheme (b). The core life for (b) was 25% greater than for (a).

Burnable poisons can also be used to flatten both local power peaks and over-all radial power peaks. For the latter the best scheme is to vary the distribution of poison (rather than fuel) to obtain a flat thermal power. Figure 3 shows results of this method obtained by KAPL for the small light-water highly enriched reactors previously described. The core is divided into two radial zones of different poison concentration. Since the over-all power distribution peaks in the center of the core, V_2, the outer zone volume, should have the lower poison concentration. The aim is to make the peaks at the core center and at the zone interface as small as possible by adjusting V_2/V_t (where V_t = total core volume) and β

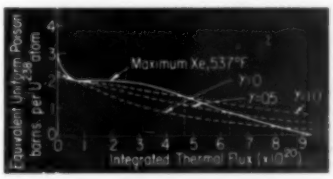


FIG. 4. Boron poison shim control in slightly enriched core

the ratio of poison density in the outer and inner zones. In general the peaks flatten rapidly for $V_2/V_t > 0.6$. A β of 0.05 gives the best pattern, but such a lightly poisoned outer zone would probably make it difficult to shut the core down. A β of 0.20 is more realistic and has only a slightly worsened pattern.

Slightly Enriched Cores

Studies of light-water moderated cores at Westinghouse show that burnable poisons can be used to advantage in cores containing slightly enriched fuel. The savings in mechanical shim control is illustrated by Fig. 4 for a 4% enriched core loaded uniformly with boron poisons of various amounts of initial self shielding. For each self-shielding parameter, γ, the boron concentration is optimized to minimize the amount of mechanical control. The maximum difference in ordinate during core lifetime between the γ-curve (dotted curves) and the "maximum Xe" curve (solid curve) indicates the amount of mechanical control needed. Compared with an unpoisoned core (equivalent to a dotted curve along the x-axis) the unshielded poison ($\gamma = 0$) gives considerable savings on mechanical shim control. Further reduction is possible with self-shielded poisons but, as shown in Fig. 4, this is only at the expense of a much shorter reactor life.

Pu Fast Reactor Performance

Both the U. S. (*637*) and Russia (*2038*)* reported on critical experiments dealing with Pu-fueled fast reactors. The American paper included a calculated performance comparison (see table) of various types of fast reactor systems having practical engineering compositions. The basic reference design for the reactors shown specified a core composition of 50% Na, 25% steel and 25% fuel and fertile material; the blanket has a composition of 20% Na, 20% steel and 60% fertile material and is 45 cm thick. These values were based on experience with EBR-II and the Enrico Fermi Reactor.

Reactor #3 is a metal-fueled design using Pu containing considerable Pu^{240}, the result of long-term irradiation in a thermal flux. The breeding ratio (based on the destruction of Pu^{239} atoms alone) is higher than for reactor #2 (with no Pu^{240}) and the critical mass is lower. This is because Pu^{240} has a higher fission probability than the U^{238} which it has partially displaced.

Reactor #4 represents one possible oxide reactor design, with a relatively low density of oxide. The median energy for all core fissions is much below that for the metal-fueled reactor, the parasitic capture rate in fuel is up, and the breeding ratio has dropped appreciably, as has the critical mass. Reactor #5 represents a second possible oxide design, this one having more fuel, and less coolant and structure (hence, lower thermal performance, other things being equal). The critical mass goes up slightly and much of the loss in breeding is recovered, although the radiative capture rate is only slightly improved. The improved breeding is due to the considerable increase in "fast fission bonus" in U^{238}. The change in leakage from the reactor is only a slight factor.

Reactor #6 is a hypothetical carbide reactor, and #7 again is the same type with an increased volume of carbide in the core. The predicted performance is very similar to the oxide reactors except for slight shifts in spectra.

Reactors #8 and #9 present two designs with thorium as the fertile material. Reactor #9, fueled with plutonium, indicates a considerable drop in conversion ratio from the Pu-U^{238} system. This is attributed primarily to the loss in fast fission bonus. In addition the thorium blanket allowed too many neutrons to escape in the design used. Reactor #8, fueled with U^{233}, again indicates an abnormally low breeding ratio due to an inadequate blanket. If one had a blanket efficiency equal to that of reactor #2, the breeding ratio would be raised to 1.27, which provides an estimate of the potential performance of this system. The low alpha for U^{233} is noteworthy. Alpha remains essentially constant over the wide range of reactor designs, including oxide and carbide systems.

While no dilute U^{233} criticals have been performed, reasonable agreement between experiment and theory for substitution experiments with U^{233} in ZPR-III has been obtained. The performance of the Pu-Th system provides the basis of an interesting speculation. One can envisage a fast-thermal partnership where the fast Pu-Th system burns plutonium and makes U^{233}, feeding the latter into a thermal U^{233}-U^{233} oxide reactor which burns U^{233} and makes Pu^{239} with a good overall neutron economy.

* Not available at press time.

Summary of Fast-Reactor Physics Parameters

Reactor No.	1		2		3		4		5◆		6		7◆		8		9	
Fuel (volume fraction)	Pu^{239} (.25)		Pu^{239} (.25)		Pu^{239}–Pu^{240} (.2) (.05)		Pu^{239} Oxide (.0248) (.083)		Pu^{239} Oxide (.0275) (.166)		Pu^{239} Carbide (.0256) (.088)		Pu^{239} Carbide (.0291) (.176)		U^{233} (.0321)		Pu^{239} (.034)	
Fertile material (core volume fraction)	U^{238} (0.0)		U^{238} (0.0)		U^{238} (0.0)		U^{238} (.0682)		U^{238} (.1386)		U^{238} (.0969)		U^{238} (.1759)		Th (.218)		Th (.2156)	
Core volume (liters)	400		800		800		800		800		800		800		800		800	
Ratio fuel atoms to diluent atoms	0.178		0.140		0.136		0.426		0.198		0.334		0.165		0.236		0.256	
Critical mass (kg)	288		468		456		378		419		390		444		475		524	
Flux integral for:	Core	Blanket	Core	Blanket	Core	Blanket	Core	Blanket	Core	Blanket	Core	Blanket	Core	Blanket	Core	Blanket	Core	Blanket
Energy interval (Mev)																		
1 2.25 – ∞	3.84	0.55	4.02	0.40	4.02	0.39	5.76	0.64	5.54	0.59	5.49	0.58	4.92	0.49	4.73	0.61	4.70	0.60
2 1.35 – 2.25	4.93	0.88	5.27	0.64	5.27	0.64	7.80	1.05	6.90	0.90	7.28	0.95	6.05	0.77	6.40	1.10	6.37	1.07
3 0.825 – 1.35	7.11	1.91	7.81	1.42	7.75	1.41	9.71	2.01	8.00	1.63	9.77	1.98	8.18	1.58	9.04	2.44	9.05	2.40
4 0.500 – 0.825	12.89	6.97	14.80	4.68	14.61	4.62	16.31	6.15	14.09	5.10	15.02	5.69	12.30	4.45	16.16	7.53	15.96	7.42
5 0.300 – 0.500	14.27	8.36	16.87	6.52	16.70	6.46	14.94	7.69	12.01	6.18	16.49	7.70	13.96	6.09	17.31	9.93	17.45	9.80
6 0.183 – 0.300	12.39	8.09	15.02	6.41	14.91	6.35	17.00	7.93	15.15	6.32	15.03	7.51	12.46	5.90	14.56	9.09	15.05	9.08
7 0.110 – 0.183	10.40	7.18	12.85	5.76	12.76	5.71	15.41	7.39	13.08	6.10	13.67	6.82	11.28	5.35	11.67	7.51	12.47	7.62
8 0.067 – 0.110	8.40	6.80	10.66	5.52	10.60	5.47	13.04	7.14	11.02	5.99	11.73	6.58	9.80	5.18	8.88	6.45	9.68	6.61
9 0.025 – 0.067	8.20	8.00	10.04	6.62	10.76	6.57	15.30	8.98	14.38	7.66	14.45	8.27	12.05	6.64	8.28	6.94	9.29	7.24
10 0.00911 – 0.025	4.20	3.58	5.07	3.06	5.81	3.08	10.85	4.65	8.77	3.93	10.25	4.29	8.11	3.34	4.34	3.37	5.22	3.64
11 0.0 – 0.00911	1.90	1.46	2.95	1.30	2.91	1.28	10.67	2.67	7.21	2.04	11.05	2.58	7.96	1.88	1.98	1.52	2.74	1.74
Fuel fissions ($X10^3$)	297	0	294	0	286	0	321	0	302	0	310	0	296	0	306	0	331	0
Fuel captures ($X10^3$)	57.6	0	60.9	0	58.8	0	82.5	0	74.2	0	82.2	0	74.4	0	32.6	0	64.1	0
Diluent fissions ($X10^3$)	45.9	20.8	50.2	15.2	62.5	15.1	19.4	24.6	42.2	21.9	24.2	22.4	47.7	18.4	8.4	3.5	8.2	3.3
Diluent captures ($X10^3$)	192	389	253	319	256	316	108	448	209	369	131	420	242	329	171	310	104	319
Na captures ($X10^3$)	1	0.3	0.3	0.2	1	0.2	2	4	1	0.3	0.2	26	1	0.2	1	0.3	1	0.3
Structural captures ($X10^3$)	15	9	19	7.3	28	7.2	28	10	14	8	0.3	9	13	7	2	9	191	9
Internal breeding ratio	0.540		0.713		0.748		0.356		0.540		0.328		0.652		0.408		0.465	
Total breeding ratio	1.64		1.61		1.63		1.37		1.52		1.38		1.54		1.15		1.27	
Ratio fuel captures to fuel fissions	0.194		0.207		0.207		0.257		0.246		0.259		0.251		0.0845		0.194	
Escape from blanket	0.0254		0.0183		0.0181		0.0239		0.0199		0.0225		0.0180		0.069*		0.069*	
Median fission energy, all core fissions (Mev)	0.367		0.334		0.305		0.181		0.234		0.197		0.269		0.251			

◆ Core Na fraction = 0.35; core steel fraction = 0.15. * A 45 cm reflector is not adequate for these two systems. Potential breeding ratios, assuming more efficient reflectors, are 1.27 & 1.40 for #8 & #9 respectively.

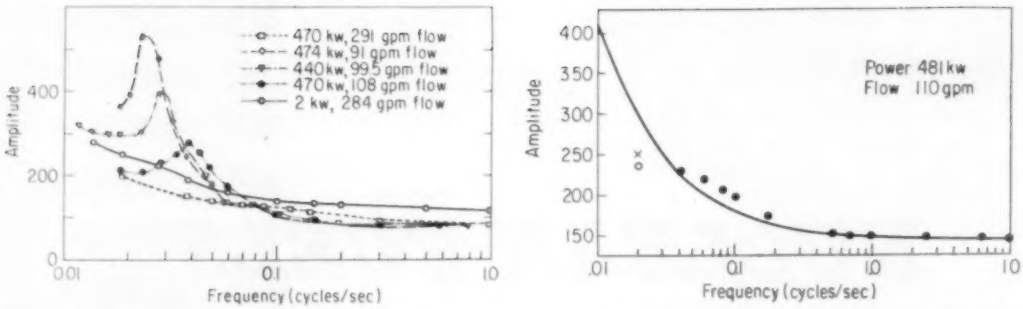


FIG. 1. Transfer-function-amplitude measurements for EBR-I with Mark-II core (left) show resonance peak at full power and low flow. Under same conditions but with Mark-III core, similar measurements (right) show no resonance peak (solid curve is theoretical zero-power amplitude). Close correspondence between points for parallel flow (crosses) and points for series flow (circles) rules out blanket-coolant temperature gradient in series flow arrangement as possible source of delayed negative temperature coefficient seen in Mark-II

Mark-III Makes EBR-I Stable

Transfer function measurements on EBR-I (*1845*) show that, with the new Mark-III core, the reactor will be extremely stable over its operating range and that the power at which instability sets in will greatly exceed the reactor heat removal capacities. (Extreme extrapolation of the available data indicates instability at 40–50 times design power at full flow.) Thus Mark-III has proved correct the contention by fast-reactor designers that EBR-I's instability troubles were a function of mechanical design and not an innate characteristic of fast reactors as a general class.

EBR-I Analysis

Analyses of the instabilities observed in EBR-I with the Mark-I and Mark-II cores (NU, April '57, 62) had led to the conclusion that the reactor possessed both a prompt positive temperature coefficient and a delayed negative coefficient (or coefficients). The source of the prompt coefficient was suspected to be bowing of the fuel element rods caused by temperature gradients in the rods.

The physical source of the large negative coefficient has remained a mystery. One of the British papers (*1462*) elaborates on a previous suggestion that the inner blanket-to-core series flow arrangement in EBR-I was responsible. They find that (a) radical power variation in the blanket, (b) heat leakage from core coolant to blanket coolant through the core skirt and (c) variation of blanket coolant-channel size would combine to cause a large steady-state radial temperature gradient in the coolant leaving the blanket. This gradient could be carried over into the core to some extent, resulting in a lower temperature on the axis than on the periphery at core inlet. Such a gradient would lead, in the steady state, to an outward bowing of the fuel elements. This bowing would be opposite in direction to the bowing caused by the instantaneous power generation pattern of the core and would have a longer time constant (i.e., the time for blanket-heating plus coolant-transient).

Regardless of their origin, the observed temperature coefficients in Mark-I and Mark-II allowed a qualitative explanation of EBR-I's resonance instability: H. Bethe showed that any reactor, if it has a delayed negative coefficient, will have a resonance at some threshold power, the power depending on the magnitude of the coefficient. The effect of the prompt positive coefficient is to lower the resonance threshold power.

Mark-III Experiments

Last fall EBR-I resumed operation with Mark-III—a core designed specifically (NU, Feb. '58, 122) to test the fuel-element-bowing and series-flow theories of temperature-coefficient origin. Tests have been made with powers up to 500 kw and flows as low as 110 gpm. Flow changes at power showed no indication of a prompt positive coefficient, at all powers up to 500 kw with flow changes from 295 gpm to 110 gpm. The conclusion is that the changes of mechanical design that eliminated fuel element bowing in Mark-III also eliminated the positive coefficient observed in Mark-I and Mark-II.

Moreover, reactor transfer function measurements showed little sign of the resonance behavior exhibited by Mark-I and Mark-II (see Fig. 1) for either series or parallel flow situations. This is taken to mean that in the absence of the prompt positive coefficient the resonance threshold has moved up in power beyond the previous operating range. Argonne feels that the complete lack of any effect on the transfer function of changing from series to parallel flow (see comparison in Fig. 1) rules out the mechanism suggested above as the origin of the negative coefficient.

On the other hand, the British authors still insist that fuel displacement of some kind must be involved; heating of reactor structure alone, they feel, could not give such a large reactivity coefficient. At this point Argonne is inclined to believe that the negative coefficient in Mark-I and Mark-II was not caused by a single mechanism but was rather the net result of the combination of many temperature-dependent effects.

Because of the reduced effect of the negative coefficient (or coefficients) with Mark-III, no additional experimental information about the origin of such coefficients has yet been obtained. Current tests at higher power should provide more definite information.

Geneva 1958

NUCLEAR FUELS

As nuclear power moves from prototype to full-scale stage, more emphasis is being placed on materials for high temperature and high burnup. The papers reported on here concern irradiation effects on fuel materials and fabrication of fuels that provide reliable, economic operation under such conditions.

United States Fuel Development

PWR Core-2 Blanket

A plate-type blanket element with UO_2 in compartmented Zircaloy-2 cladding (see Fig. 1) is under development for the second core of the Shippingport reactor (*2380*). Benefits of this design will be (a) higher thermal performance [100 Mw(e) vs the present 60 Mw(e)], (b) increased life (10,000 full-power hours for the first seed), (c) lowered fabrication costs.

The fuel is cold-pressed and sintered to a density of >96% of theoretical. This density, higher than for core 1 (93–95%), has better fission-gas retention. Also, the grinding operation used for core 1 pellets is eliminated—fuel thickness can be maintained within ±0.002 in. by pressing and sintering. Length and width are not critical since ~5% void space is desirable in each fuel compartment. The fuel is not bonded to the cladding, necessitating compartmentation of the cladding to provide rigidity.

A new procedure, eutectic-diffusion-bonding, was developed for bonding the cladding components together. It is relatively simple, inexpensive and provides strength, ductility and leaktight compartments. The bonding involves formation of a molten eutectic alloy film between the cladding-component surfaces. The film is formed by diffusion of a thin layer of pure metal on the surfaces of the fuel-receptacle plate during heating of the assembled components to the bonding temperature. On reaching the melting point of the eutectic alloy, the latter melts and bonds the higher-melting-point cladding components. Holding the assembly at the bonding temperature for an extended period results in diffusion of the bonding metal and leaves no solid-eutectic-alloy layer in the bond interface as in brazing.

Nickel, Cu and Fe were investigated as bonding agents. Their lowest-melting-point eutectics with Zr melt at 961, 885 and 934° C, respectively (Zr melting point is ~1,750°). Bonding-agent layers of 0.15–0.25 mil are obtained by a chemical-displacement technique—Zr displaces Ni, Cu or Fe from salts in an acid bath (room temperature for Cu, 90–100° C for Ni and Fe) to provide a bonding-agent layer on the cladding surface. Thickness of the layer can be controlled to ±0.02 mil. This method is preferred to electroplating because of simplicity, no hydrogen evolution or contamination and uniform coating over geometry variations.

Intimate contact of the surfaces during bonding is ensured by fusion welding the outside periphery and by an external inert-gas pressure. A bonding temperature of about 1,000° C is applied for several minutes to several hours. Fast heating and cooling rates are required to avoid, respectively, excessive diffusion of the bonding agent before reaching the eutectic melting

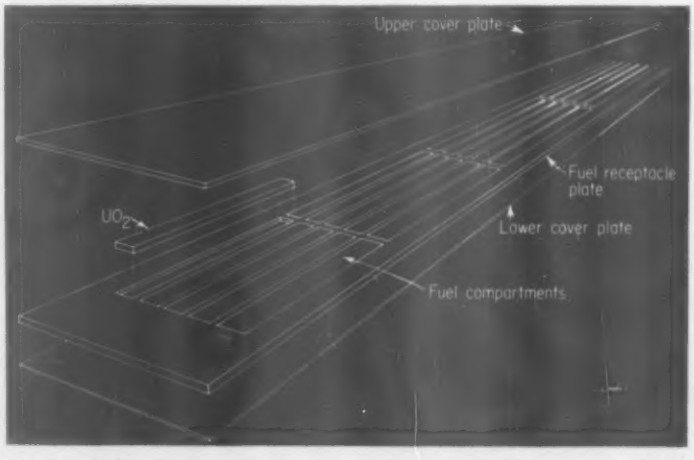


FIG. 1. Blanket element under development for second PWR core—sintered UO_2 in compartmented Zircaloy cladding

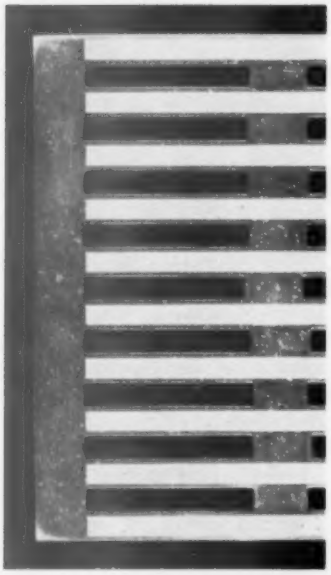


FIG. 2. Gas-pressure-bonded Zr-clad Zr-U assembly bonded at 10,000 psi for 5 hr at 1,550° F

point and precipitation of alloying constituents in the cladding.

Out-of-pile testing has shown no thermal-cycling dimensional-change problems and adequate strength and ductility. Copper and iron do not adversely affect corrosion resistance. However, nickel is unsatisfactory because it results in excessive absorption of hydrogen and embrittlement of the cladding during corrosion testing.

Th-U Elements for SRE

The Sodium Reactor Experiment (*785*) includes thorium-rich, thorium-uranium alloys as part of its program to develop fuel materials for long burnup at high temperature.

Thorium has promising properties for this type of service—face-centered-cubic crystal structure, high melting point (1,700° C) and no phase changes below 1,400° C. However, there is a lack of high-temperature irradiation data.

The Th-U irradiation slugs are 0.75-in. diameter and 6.00 in. long. The composition is 7.6 wt% highly enriched uranium. Density is 11.9 gm/cm³. Peak fuel temperature will be 650° C at the center and 510° C at the surface. As the data is developed, this temperature may be raised. The alloy composition is not representative of a large-scale power plant; in the latter case, 3–5% uranium would be used.

The solubility of uranium in thorium at room temperature is <1% and increases to ~1.5% at 600° C. The excess uranium is homogeneously dispersed as a finely divided second phase. The alloy is prepared by vacuum-induction melting in graphite crucibles. Only the tapered bottom of the mold is coated with BeO or magnesium zirconate; the tapered design provides a leading shape for extrusion. A homogeneous ingot is produced by pouring at a minimum superheat into a cold mold. Because of migration of radium isotopes and other daughter products of Th^{232} during melt, the ingot surface is highly radioactive. Special handling precautions include gloves and respirators.

The machined ingot is placed in a 16-gage-wall copper can to minimize oxidation during preheat and extrusion. The canned billet is given an extrusion preheat of 870° C for 3–4 hr and then extruded in a 1,000-ton press through a 0.850-in. die preheated to 260° C. The reduction ratio is 30. The rod is then swaged to 0.750-in. diameter and cut to 6-in. slugs.

Direct vacuum-casting of the slugs is being considered as a future low-cost fabrication method. However, major problems are alloy homogeneity and mold materials.

UO_2-SS Fabrication

Powder rolling has been found to provide UO_2-stainless steel dispersions with good quality control and fairly low production costs using standard fabrication equipment (*784*). Although developed for thin-walled tubes (0.0175–0.0185 in. thick), the technique can be applied to plates and other geometries. It is possible to form higher UO_2 percentages than are common to plate-type elements.

The nominal composition of the tube cermet is 25% UO_2 and 0.1% boron as B_4C. High-fired UO_2, of particle size 44–88 microns, is used since other oxide particles did not stand up under the fabrication.

The stainless powder is commercial-grade high-silicon 304. Atomized stainless is selected to ensure chemical composition and to limit particle size.

Close control of powder characteristics is essential since the thickness uniformity required within the green rolled strip is ±0.001 in. The powder is blended in a 1-quart "L" pack blender for ~1 hr. Density varies 2.86–3.00 gm/cm³ and flow rate varied 23.40–26.50 sec/50 gm in 107 lots.

Following rolling, the strip is sintered and rolled to full density. The strip is then sheared and machined to close tolerance to assure bonding after tube forming. The surface is chemically cleaned to permit good metal-to-metal contact with the cladding and to remove UO_2 particles that might penetrate cladding.

Gas-Pressure Bonding

Developed for cladding fuel elements, gas-pressure bonding (*788*) entails sealing the parts to be bonded in an envelope, which is evacuated and heated to the bonding temperature in gas at 700-

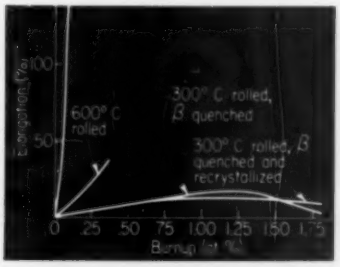


FIG. 3. Uranium metal elongation under irradiation as influenced by fabrication history

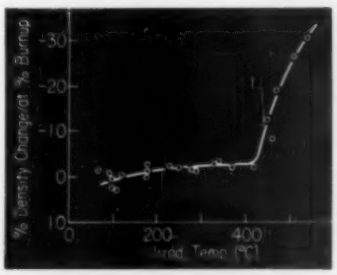


FIG. 4. Effect of irradiation temperature on swelling of uranium compacts prepared by hot-pressing uranium hydride powder

atmosphere pressure. The method has the advantages:

1. Complex shapes are easily produced.
2. Brittle materials can be clad.
3. Close dimensional control can be achieved.
4. Cost is low.

The bonding temperature must be high enough so that plastic flow of the materials occurs rapidly (850° C for Zr alloys). Surface preparation is important—pickled and grit-blasted Zircaloy

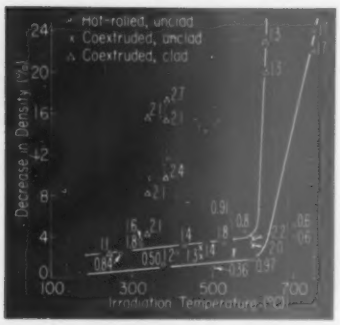


FIG. 5. Irradiation effect on density of U-10wt%Mo (fuel for Enrico Fermi reactor) as function of irradiation temperature and burnup

surfaces are slightly contaminated and bond poorly; grain growth does not occur between cold-rolled surfaces; machined or abraded Zircaloy bonds perfectly.

Other than zirconium, good bonds have been achieved with aluminum, stainless steel, molybdenum and niobium clads. They have been applied to cermets and uranium dioxide.

Dispersions have been clad without the severe stringering encountered in conventional techniques. Materials handled have been Ti-B clad with titanium and U-alloy–UO_2 clad with molybdenum and niobium.

Gas-pressure bonding is most promising for bonding and cladding subassemblies in one operation. An example is shown in Fig. 2. The assemblies are as strong as those prepared by other techniques and far more accurate. The process also promises to be much less expensive.

Irradiation Effects

Irradiation investigations of fuels at Argonne National Laboratory (*1890*) have emphasized high burnups at elevated temperatures, as would be necessary for low fuel costs for civilian power reactors.

Most irradiations have been made on cylindrical specimens 1.00 in. long with 0.125–0.250-in. diameters, usually in NaK capsules. In the case of thermal-reactor fuels, attention has been mainly confined to uranium that is unalloyed or has small alloy additions.

Unalloyed uranium. Figure 3 shows the dependence of irradiation effects on the fabrication history of wrought uranium. Quenching decreases lengthwise growth considerably, but considerable surface roughening develops. Recrystallized specimens display essentially the same behavior as unrecrystallized specimens. Specimens rolled at 600° C display intermediate effects. Density changes are essentially independent of fabrication history—initially, density decreases ~4.8% per at% burnup; after 0.5 at%, the rate decreases so that at 1.8 at% burnup density has decreased only about 3.9%.

Irradiations were made on compacts prepared by hot-pressing uranium and uranium-hydride powders. Under irradiation, full-density and less than full-density uranium-powder compacts shortened in the direction of pressing. Specimens prepared from uranium hydride all elongated under irradiation in the pressing direction; as shown in Fig. 4, density changes were dependent on irradiation temperature. High volume increases were the result of central void formation and a surrounding zone of porosity—probably the result of nucleation and growth of bubbles of volatile fission products.

Fermi reactor fuel tests. Previous irradiation tests had shown U-10wt%Mo to be a promising fuel for the Enrico Fermi fast breeder reactor. Subsequently, the material was submitted to higher temperatures and burnups (*622*).

Tests of burnup up to 2.5 at% showed good irradiation stability of gamma-treated alloy to 2 at% burnup below 600° ±90° C. Large changes, such as swelling or rupture, occurred at burnups <0.8 at% at 700° C. Ruptures occurred in material restrained by 8-mil cladding or fabricated improperly. Figure 5 shows density effects.

Physical property changes were related to the initial condition of the material, total burnup and in-pile temperature. Marked reductions in ductility and strength were noted, although the part played by temperature is not clear.

Irradiation effects in UO_2. Irradiations were carried out by Knolls Atomic Power Laboratory of UO_2 powders produced by low-temperature steam oxidation of U metal and of high-density UO_2 dispersed in stainless steel (*615*).

The powders, put into the reactor at 60% of theoretical density, sintered to form a porous mass. The fractured dispersion particles also sintered. The sintering rate under irradiation was much greater than temperature alone could have caused.

The UO_2 particles also become quite porous during irradiation. This is caused by fission-gas agglomeration and is affected by burnup, irradiation temperature, time, restraining pressure and fission density. The porosity becomes very prominent at 10–20% burnup, regardless of other conditions.

Figure 6 (left) shows a UO_2 particle, irradiated at 1,000°–1,500° C, that was completely unconfined due to fuel-element failure. The UO_2 has undergone gross swelling without fracture because of irradiation-induced plasticity.

For low-temperature irradiations, pores are small and uniform in size and distribution. At higher temperatures, pores are larger, poorly distributed and of varying size. The porosity of unconfined particles is similar to that for high-temperature irradiation; Fig. 6 (right) is a good example of this where differences in porosity are caused by variations in restraining pressure.

High fission density also causes pronounced plastic deformation of UO_2. Figure 6 shows such flow where a particle is confined on all sides but one by the stainless steel matrix.

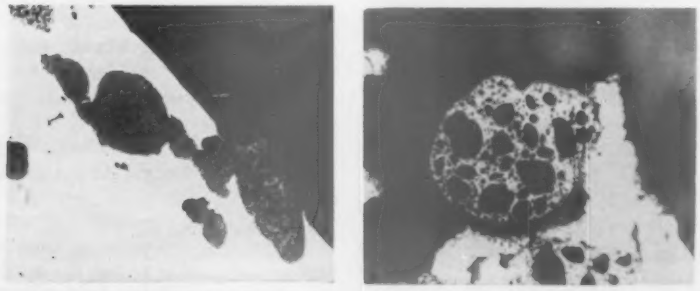


FIG. 6. Porosity growth in UO_2 under irradiation. Particle at left was unconfined when element ruptured at 1,000–1,500° C; burnup was 12 at%. At right, unclad UO_2–stainless-steel dispersion irradiated to 19 at% burnup at 300° C

France Plans Fuel Modifications

Paralleling British development, French efforts are concentrated on clad uranium elements for gas-cooled reactors.

The table compares fuel element characteristics of the current type (Mg-Zr, longitudinal-fin tube) with those of a contemplated type for pressurized-gas reactors (*1157*).

The contemplated fuel elements may use beryllium or possibly zirconium alloys as jacket materials. Beryllium welds without difficulty and withstands CO_2 corrosion at 600–650° C. Its shaping as finned tubes currently appears impossible. Thus, the use of beryllium leads to broken-up ("bunch") types of fuel elements.

Uranium-molybdenum (1–1.5%) alloys appear most promising. For a jacket temperature of 500° C, the bunch shape of fuel elements makes it possible not to exceed a hot spot of 600° C in the fuel metal.

Automatic Welding

The French have developed automatic welding procedures (*1161*) that provide fuel-element end welds with good mechanical properties and corrosion resistance. The techniques—argon arc, inert-atmosphere argon arc and electronic bombardment—are applicable to uranium, magnesium, aluminum, beryllium, stainless steel, zirconium and refractory metals.

Argon arc. This method is used for sealing the magnesium-alloy-sheathed elements of G2. The unit consists of a commercial argon-arc generator, arc extinguisher, cam-operated programmer and a machine controlling the positioning and rotation of parts.

Except for inserting the element to be welded in the chuck and engaging the automatic device, the welding cycle is automatic. This technique can be used for aluminum and stainless steel, but it is unsuitable for oxidizable metals such as uranium and zirconium.

Inert atmosphere. Argon-arc welding in an argon atmosphere is suitable for uranium and zirconium, although electronic bombardment is preferable for the latter to maintain purity.

Electron bombardment. Electron-bombardment welding is performed in a vacuum of 10^{-5} mm Hg by focusing an electron beam on the part. It is especially attractive for metals sensitive to gas contamination.

Welds obtained with zirconium and Zircaloy show good ductility and homogeneity without blister, crater or crack. Uranium and molybdenum have been welded successfully.

Fuel for Gas Reactors (*1157*)

Parameter	*Current fuel types*	*Contemplated types*
Jacket hot spot (°C)	≤400	500–650
Max. power density (watts/cm²)	80–100	150–200
Burnup rate (Mwd/tonne):		
U, 15% Mo	1,000–1,300	
U alloys, refractories		≥3,000

Germany Uses New Fuel Type

An interesting German project is the pebble-bed reactor; the fuel is described in this section (an over-all description of the reactor is given on p. 138). Also noteworthy is a technique for bonding aluminum cladding to uranium-niobium alloy.

Pebble-Bed Reactor Fuel

Uranium carbide in spherical graphite bodies has been adopted to fuel the German (AVR) rare-gas-cooled, high-temperature pebble-bed reactor (*1005*).

The virtues of this fuel form are mechanical strength, resistance to thermal shock and radiation damage and ease of charging and discharging the reactor. The fuel must operate at a maximum surface temperature of 1,100° C.

The graphite bodies have a diameter of 5–6 cm. The fuel, 20%-enriched UC mixed with carbon, is inserted into a borehole in the sphere. The borehole is not completely filled to avoid mechanical stress. It is hermetically sealed by a plug cemented into place.

Synthetic reactor graphite will be used. Required mechanical crushing strength is 200 kg/cm². Specific surface area of the spheres is 100–300 cm²/gm fissionable material.

Specific output of the fuel is set by the maximum possible fuel temperature, which is a function of the fuel vapor pressure. An initial specific output of 2 kw/gm of fissionable material should be readily attainable—the maximum calculated output is many times higher.

The fuel bodies should withstand a dose of at least 5×10^{21} nvt. Since even the coldest fuel bodies will be at 300° C, Wigner energy can be neglected.

Contamination of the gaseous coolant must be tolerated because of diffusion through the graphite. Chief contaminants will be the gaseous fission products Xe and Kr. Most other fission products form stable carbides, which, like UC, have a low diffusion rate in graphite. Among the non-gaseous fission products, Te, Se, I and Br form no stable carbides and diffuse at an appreciable rate.

New Bonding Method

Although the Al-canned, natural-U fuel elements for the Karlsruhe research reactor (FR-2) are a conventional research-reactor type, a can-to-core bond was developed offering two advantages—it protects against uranium corrosion and prevents uranium diffusion into the aluminum (*1004*).

The 216-cm-long, 32-mm-diameter U-1%Nb fuel cores are canned in 2S-aluminum. The Al can is forced over the core rods, welded and bonded to the rods in a gas autoclave at 380° C and 150–200 atmospheres.

The bonding metals are Zn and Sn. After surface-cleaning the uranium rods, an 8–10 μ Zn layer and then a 12–15 μ Sn layer are vacuum-evaporated very uniformly onto the surface.

When drawing the Al can onto the rods, deformation splits the oxide layer on the Al tube to permit bonding. In the autoclaving, the Sn-Zn eutectic forms around 200° C. At 350° C, UZn intermetallic and a mixed crystal of Sn and Al form.

The shearing strength of the bond is ~2.7 kg/mm². In a test of 1,000 hr at 100° C, no changes in the bond were detected. At 350° C, there was slight weakening; at 450° C, there was bond separation.

United Kingdom Gives Fuel Experience

A noteworthy aspect of British design (*73*) is the difference in approach between stacked fuel elements (Calder Hall, Bradwell) and individually supported elements (Hunterston). Stacking offers simplicity but subjects the elements to strain. Independent support involves additional complications in manufacturing and charging and discharging—additional life of 200–300 Mwd/tonne is necessary to justify the extra cost.

In-Pile Experience

During the past five years, irradiated fuel elements have been examined from Windscale and Calder Hall (*50*).

Since startup, fuel element failures at Windscale have been <0.1%. At Calder Hall, only 13 have failed or been suspected of failure out of ~30,000. Other than failures, irradiated elements have been examined for surface wrinkling, metallographic structural changes, density and dimensional changes and creep.

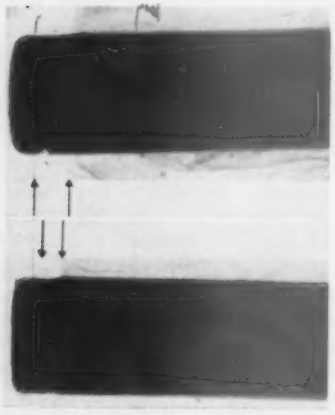


FIG. 1. One type of Windscale fuel-element failure—endcap rupture caused by can weakness and irradiation growth; radiograph of two ends shows extension at one end

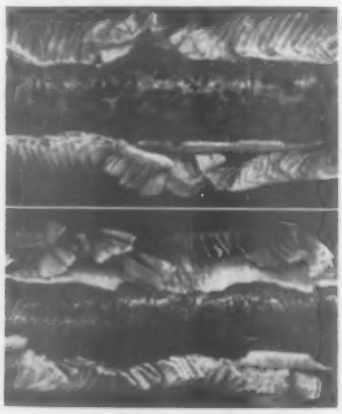


FIG. 2. Inside surfaces of cans cut from Calder Hall elements. Top can is from sound element; bottom one is from defective element in which oxide patches are evident

Failures. Some fuel-element failures are shown in Figs. 1 and 2. Most of the information on failures has come from the study of Windscale elements. The failures fall into the types:

1. An early type of failure was a circumferential tear around the end of the can due to large wrinkles. This involved over-all core-length increases of up to 0.3 in.

2. Fabrication of one group of elements caused thinning of the can wall near the endcap. Strains arising during a comparatively short irradiation period were sufficient to cause fracture.

3. A break in the medium separating the Al can from the uranium permitted formation of UAl_3 with resulting volume increase and embrittlement.

4. In a batch of elements fabricated by a particular technique, irradiation growth plus a local weakness in the can caused rupture (Fig. 1).

In the case of Calder Hall, pinpointing the failed cartridge among the six discharged from a channel is difficult. Radiography is successful if oxidation has changed the core profile. In other cases, all the cartridges were decanned and the inside of each can inspected (Fig. 2). In these cases, the failure-detector signal rose slowly above background.

In several cases, failure was due to a poor weld at the endcap. In the remainder, neither can nor fuel appeared to suffer any significant change. Most failures have been attributed to fabrication defects.

Wrinkling. Some early Windscale elements of cast material developed an irregular wrinkled appearance after irradiation. This has been related to growth in large-grain-size uranium. It has been corrected by beta quenching. No wrinkling has been observed in Calder Hall elements, which are heat treated.

Density. Small density decreases have occurred in Windscale elements. The effect is greatest at 150–200° C (0.7% for 1,000 Mwd/tonne and 220° C). Irregular holes, 10^{-2}–10^{-4} mm² and frequently associated with inclusions, are believed related to irradiation growth.

Dimensions. Length changes in cast-and-machined bars have only occasionally been above 1% for 1,000 Mwd/tonne—usually a length increase.

Creep. In the Calder Hall reactors six elements, each weighing 14 kg, are stacked vertically in each channel. Radial arms provide lateral support at the top of each element. Mean can-surface temperatures range from 250° C for the bottom element to 400° C for the element next to the top. During initial reactor operation it was found that for an element with an initial bow of 0.025 in., a bow of 1 in. would develop after ~550 Mwd/tonne. As a remedy, a magnesium-alloy brace was attached to the middle of the can. No observations of its effectiveness have yet been made at high irradiations.

Irradiated-U Swelling

A problem that will assume even greater significance for British reactors at necessarily higher burnups is swelling of metallic uranium. The important mechanisms leading to swelling are recognized, but a complete quantitative description cannot yet be given (*81*).

Volume increase due to pressure exerted by bubbles of fission product gas has been calculated on the basis of a number of assumptions. Several conclusions have been drawn regarding the mechanism. At high temperatures and low irradiation rates, creep is dominant. At low temperatures and high irradiation rates, swelling is smaller for a given burnup; plastic flow then becomes a significant factor. However, for uranium to become entirely plastic, a burnup rate of 1% in a few weeks would be required; thus, creep is always likely to be important.

Figure 3 shows uranium swelling calculated for two irradiation rates from measured creep-strength data. The 2% inclusions by volume assume normal British uranium (2% non-

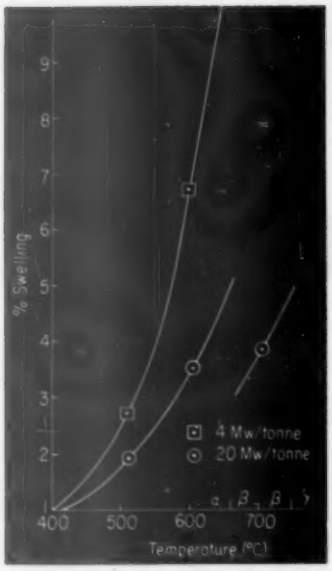


FIG. 3. Swelling calculated from uranium-metal tensile creep data for 0.3% burnup

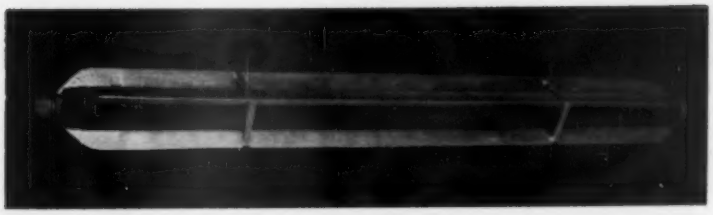


FIG. 4. Bradwell polyzonal spiral-finned fuel element

metallic inclusions and 0.8% intermetallic particles). As indicated by the 4 Mw/tonne curve, a typical gas-cooled-reactor rating, swelling is greater at lower rates of irradiation.

The calculated results show that the bubble mechanism plays no part in swelling at temperatures below 400° C. However, large volume increases are predicted at 600° C and above, in agreement with observed results.

Spiral-Finned Fuel

A significant increase in power output per unit core volume over Calder Hall has been achieved for gas-cooled industrial power reactors under construction, e.g., Bradwell (Fig. 4), principally by a polyzonal-spiral-finned fuel element (*48*).

Spiral-fin design consists of a smooth-bore barrel with helical fins slotted to accommodate longitudinal fins (splitters) extending the length of the element. The helical fins are of small enough diameter to leave an annulus between their outer edge and the fuel-channel wall. The splitters extend from the barrel close to the channel wall. Thus the gas flow about the element is divided into four zones in each of which the gas follows a helical path. This provides good heat transfer by continuously replacing gas in the helical grooves with gas from the main stream.

Bradwell Power Station

The Bradwell reactors (*263*) use the spiral-and-splitter fin design (Fig. 4). (Over-all description of Bradwell is given on p. 132.)

The higher heat-transfer coefficient of the fin design has made it economically possible to use a uniform channel diameter throughout the core and to raise the outlet-gas temperature. The spiral fins accommodate uranium growth, and the splitters prevent bowing of the lower stacked elements.

The natural-uranium-metal rods are 1.155-in. diameter and 36 in. long. The finned cans are magnesium alloy. The elements are stacked eight to a channel. The initial mean irradiation will be 3,000 Mwd/tonne.

Hunterston Fuel Element

Although the Hunterston reactors (described on p. 134) are direct descendents of the Calder Hall units, design changes in the natural-uranium, magnesium-alloy-canned fuel elements (*1523*) contribute to the more efficient operation of Hunterston, as indicated in the table.

Problems imposed by these operating conditions are bowing of stacked elements, uranium swelling and oxidation and creep of the magnesium.

Bowing is prevented by supporting the elements individually in graphite sleeves by means of beam supports. To control gas flow in the annulus between the sleeves and moderator blocks, the sleeve-graphite porosity will be reduced to $\frac{1}{50}$ that of reactor-grade graphite.

Fuel swelling will probably be the limiting factor on fuel-element life. Swelling should not exceed ~15 vol% after 4,500 Mwd/tonne at the highest operating temperature. This can probably be accommodated by the can.

Improved Design—Higher Efficiency

	Calder Hall	*Hunterston*	*Bradwell*	*Hinkley Point*
Max. can temp. (°C)	408	454	420–450	430
Max. U temp. (°C)	480	569		600
Gas outlet temp. (°C)	336	391	390	373.3
Gas inlet temp. (°C)	140	204	180	178.9
Avg. life (yr)	<1	5		
Max. burnup (Mwd/tonne)	<1,500	4,500	>3,000	3,000

FIG. 5. Hunterston fuel element is individually supported in graphite sleeve. Sleeve and element discharge as unit

Geneva 1958

FUEL CYCLES

Pu recycling will not be economically beneficial in slightly enriched light-water reactors. The economic future of the U^{235}–Pu and the Th–U^{233} cycles for other reactor types, however, looks more promising. These are the conclusions drawn from a number of studies by U. S. authors on recycling operations

PWR Survey

The most ambitious study in terms of numbers of fuel-cycle concepts studied was reported by T. H. Pigford et al (*1016*). They considered for a single region pressurized water reactor all the possible combinations of three sets of alternatives: H_2O or D_2O, U^{238} or Th fertile material, once-through or recycle fuel-flow scheme. The reference design was a 480 Mw(th) reactor with 22,000–33,000 kg of fertile material. The light-water cases used mixed oxides of fuel and fertile material clad in stainless steel. The D_2O designs had Zr-clad U metal or U^{235}-Th alloy.

The reactivity lifetimes of the four single cycle systems are plotted as a function of fuel enrichment in Fig. 1.

H_2O-U^{238} design. For once-through operation, the H_2O reactor with U^{238} feed was found capable of quite large reactivity lifetimes for fuel enrichment in the range 2.28–8%. Economic considerations predict an optimum lifetime of around 30,000 Mwd/ton (at an enrichment of 4.5%). For these conditions the net fuel cost is estimated to be 2.6 mills/kwh with Pu at $12/gm or 1.4 mills/kwh with Pu at $30/gm.

If the Pu is recycled in this reactor with enough U^{235} makeup added to each cycle to give a 6,000 Mwd/ton cycle time (initial enrichment 2.6%), the U^{235} requirement would be reduced 9% for the first recycle. At the end of the tenth recycle, the U^{235} and Pu concentrations would have leveled off to about 1.5% and 1.0% respectively.

Fuel costs for Pu recycling at 6,000 Mwd/ton would be about 1 mill/kwh higher than the once through fuel cost with Pu $12/gm. This difference is due mostly to the factor of two larger fabrication charge assumed for recycled fuel. At 32,140 Mwd/ton burnups, recycling breaks even with the once through $12/gm operation, in spite of the fabrication charge penalty.

D_2O-U^{235} design. Although enrichment considerably extends the lifetime of this reactor (see Fig. 1), the authors found that natural U was the best fuel, because of the low feed-preparation cost. In the recycle operation considered all the Pu from the previous run was fed back with the natural U feed for the next run. As the Pu built up the lifetime increased from the initial value of 5,000 Mwd/ton to over 13,000 Mwd/ton at the end of the fifth cycle.

The minimum once-through fuel cost for Pu at $12/gm would be 1.7 mills/kwh, 0 8 mills/kwh lower than the light water case The D_2O design however would incur an additional expense of 0.6 mills/kwh for heavy water inventory and makeup.

The fuel cost for this reactor with Pu recycling is slightly (~0.2 mills) smaller than the 1.7 mills quoted for once-through operation.

H_2O-Th design. The reactivity life as a function of enrichment behaves similarly to the U-fueled H_2O design. On a once-through basis, the minimum fuel cost, which occurs at burnups of 30,000–40,000 Mwd/ton, is 3.5 mills/kwh (including a U credit of $10/gm).

The method of U^{235}-U^{233} recycling studied was to enrich the Th feed with the recovered uranium and enough makeup U^{235} to give the same initial

Experiments Check High Burnup Calculations

The Canadian group (*205*) reports on experimental irradiation of natural U to check the validity of long term reactivity calculations. (Such calculations provide estimates of reactivity lifetime by using basic nuclear data to calculate changes in core composition.) Fuel specimens irradiated as high as 6.000 Mwd/tonne (5,400 Mwd/ton) were subjected to chemical and isotopic analysis and reactivity measurements in a pile oscillator. The resulting data allows predictions of the neutron balance in natural U for irradiations up to 8,000 Mwd/tonne.

The results were compared with predictions based on detailed calculations of neutron absorption in an NRX rod, taking into account time variations in the flux distribution and spatial composition.

The calculated and experimental values for the concentrations of Pu^{239}, Pu^{240}, Pu^{241} and Pu^{242} at high burnup agreed to within 1%. However, the experimental and analytical predictions of the net effective absorption cross section for the highly irradiated uranium differed by 30% (the calculations underestimate the reactivity lifetime.) Agreement with the experimental data can be forced by changing the fission-cross-section and ν ratios for Pu^{239} and U^{235} a few per cent. The Canadian work suggests that small errors may be present in the world-average values of these parameters.

excess reactivity at the beginning of each cycle. Although the amount of U^{235} makeup is considerably reduced after a number of cycles by U^{233} build-up, the lifetime per cycle is also reduced. As a result for this mode of recycling the fuel costs turn out immoderately high. A possible criticism of this finding is that the initial cycle time chosen (5,000 Mwd/ton) may be too short to be economically realistic; investigations by B & W (see below) for a zirconium-clad H_2O-Th system recycled at high reactivity lifetimes are more encouraging.

D_2O-Th design. Compared with the other D_2O reactor, the attainable burnup afforded by additional enrichment is much greater. The minimum once-through fuel cost for this design occurs at 31,000 Mwd/ton and is 1.5 mills/kwh; this compares favorably with the 1.7 mills quoted for the U fueled D_2O reactor at 11,500 Mwd/ton.

The authors found that, for an initial life of 11,060 Mwd/ton, the recovered U^{234}-U^{233} could be recycled with the Th feed without adding makeup U^{235}. Since the conversion ratio is somewhat less than 1.0, however, the reactivity life falls sharply with successive cycles. At the fifth cycle, for instance, the reactivity is less than 2,000 Mwd/ton. The authors conclude that the natural-U-fueled D_2O system is the only one of the systems studied that can operate without enriched uranium feed.

Light-Water Recycling

Impressively detailed physics calculations by the General Electric Co. (*2167*) found, in agreement with Pigford et al, that Pu recycling would bring no economic advantage (at today's fuel prices) for low enriched light-water reactors.

The G.E. paper pointed out that previous studies which supported more optimistic conclusions did not properly take into account the resonance region effects important in close-packed light-water lattices. The authors feel it is necessary in evaluating effective cross sections, for instance, to consider the changes in geometric shielding and thermal neutron spectrum produced by different isotopic compositions. Although satisfied that their treatment has eliminated all major oversimplifications, they point out that the results are subject to large uncertainties because of the basic nuclear input data. (A 10% error in α for Pu^{241} would change the reactivity by 2%.)

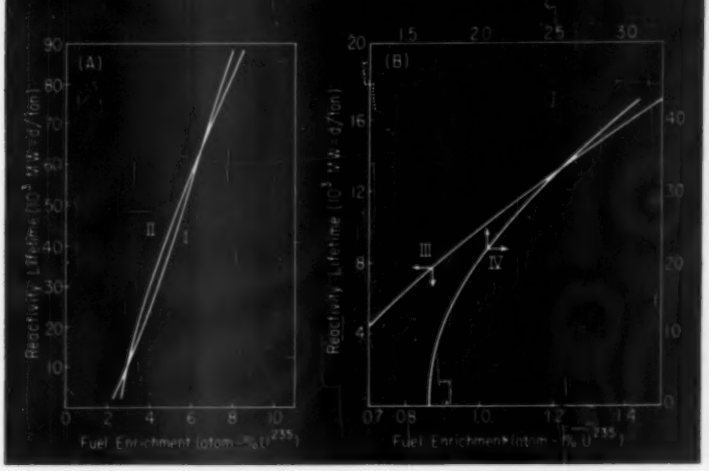


FIG. 1. Reactivity lifetime vs. fuel enrichment for four different types of pressurized water reactors. Curves I and II have an H_2O moderator and use U^{235} and Th respectively as the fertile materials. For curves III and IV the moderator is D_2O and the respective fertile materials are U^{233} and Th

The authors followed the composition and reactivity history for reactors covering a range of light water lattices. With a physical plant lifetime of 80,000–100,000 Mwd/ton and a recycling time of 4,000–8,000 Mwd/ton, a reactor would be recycled 10–25 times during its existence. Characteristically the reactivity reaches a constant value after 20,000 Mwd/ton and the core isotopic composition approaches steady-state values after 40,000 Mwd/ton.

The survey calculation results were used to make economic comparisons for a specific boiling water reactor design [630 Mw(th); 1,015 psi; 2.0 water fuel ratio; Zr clad UO_2-PuO_2]. Neglecting any cost penalties for handling and fabricating radioactive Pu, the authors found that the most economic recycling scheme would use enriched U makeup along with the Pu feed; the amount of enriched required, however, would be cut by a factor of five from nonrecycle operation. Even for this optimum scheme, the total fuel-cycle cost would be slightly greater than the cost for nonrecycle operation with the Pu sold at $12/gm. Both modes of operation appear capable of fuel costs below 2 mills/kwh, based on projected fabrication and processing costs. Recycling on natural U makeup alone, a cycle of great interest to foreign programs, appears limited to cycle times of 4,000 Mwd/ton and has fuel costs double the slightly enriched scheme.

H_2O-Th Experiments

Experiments and calculations by the Babcock & Wilcox Co. (*2405*) have yielded estimates for the reactivity life and conversion ratio of pressurized water UO_2-ThO_2 cores. The authors conclude that recycling of the U isotopes (with fission products removed) in a zirconium-clad thorium core will give an average conversion ratio of about 0.8 for 18,000 Mwd/ton exposure after three recycles. The U^{235} makeup per core loading will come to 70 kg or

Starting a Th-U^{233} Cycle With 20% U

A strategy for switching a single region graphite reactor, such as West Germany's high-temperature pebble-bed design (see p. 138), from 20% U fuel to a Th-U^{233} breeding cycle has been described (*961*).

The reactor would begin operation as a converter with a fuel charge containing separate Th and 20% U graphite fuel pebbles. Provided the Th ratio can be made high enough for a reasonable core size and power density, most of the neutrons will be captured by Th to produce U^{233}, rather than by U^{238} to produce unwanted Pu^{239}. A one-group parametric study showed that reactor physics and engineering conditions could be satisfied for Th/U^{233} ratios as high as 5–10. With a ratio of this size the author believes that the transition to a Th-U^{233} breeding cycle would be possible within a reasonable length of time. An exact estimate of the time needed for a particular reactor design depends strongly on the influence of fuel element self-shielding factors on the allowable Th/U^{233} ratios.

Properties of Plutonium Sub Infinity

Pu_∞ has been suggested as a symbol for steady-state mixture of Pu isotopes produced when Pu is exposed for a very long period to a thermal neutron flux with fresh Pu^{239} continually added to compensate for burnup. Using the following isotopic mole fractions:

$Pu^{239} = 0.217$ $Pu^{240} = 0.139$ $Pu^{241} = 0.051$ $Pu^{242} = 0.593$

Brookhaven National Lab (*461*) has calculated effective values of the basic nuclear constants for Pu_∞ as follows:

$\bar{\nu}_f = 230$ b $\bar{\sigma}_c = 184$ b $\bar{\sigma}_a = 414$ b $\bar{\nu} = 2.89$ $\bar{\eta} = 1.61$
$\bar{\alpha} = 0.80$ $\bar{\lambda} = 8.69 \times 10^{-11}$ sec^{-1} $\bar{t}_{1/2} = 253$ yr

0.23 gms per Mwd of heat (or about 0.7 mills/kwh).

The predictions were based on data from several enriched uranium-thorium-water critical assemblies. Infinite medium lattice parameters were measured for cores containing thorium metal and for cores containing thorium oxide. The oxide cores used pin-type elements containing pellets of highly enriched uranium oxide-thorium oxide fuel; the cladding was stainless steel.

The reactivity life of an oxide thorium converter can be twice as long as the life of a U^{235} converter having the same initial excess reactivity. Three factors contribute to the longer reactivity life of thorium:

- The thorium conversion ratio is higher by as much as 0.2.
- U^{233} has a larger η than Pu^{239}.
- More fuel is required in the thorium core to give the same excess reactivity.

As an example, a zirconium clad thorium core with an enrichment of 3.3% would have an initial 15% k_{ex} and would run for 20,000 Mwd/ton; a U^{233} core of the same life would require a fuel enrichment of 2.7% and would have a 36% k_{ex}. Partial reloading of the core will reduce the excess reactivity that has to be controlled. For instance, partial reloading of ⅓ of the core at a time cuts the maximum k_{ex} requirement to about 6% for thorium and 18% for U^{235}. The table shows the material balance for the first recycle for an 18,000 Mwd/ton Zr-clad core.

Materials for H_2O-Th Core

	Initial Core		*First Recycle*	
CR_{avg} →	0.68		0.73	
Isotope	*Initial (kg)*	*Final (kg)*	*Initial (kg)*	*Final (kg)*
U^{233}	0	152	170	211
U^{234}	0	12	12	31
U^{235}	476	177	272	104
U^{236}	0	56	56	83
Pa^{233}	0	18	1	18

Calder Hall Recycle

Reactivity lifetime calculations reported by American Standard (*1044*) indicate the Pu recycling in a Calder-Hall-type reactor will compare favorably with selling the Pu at $12/gm if Pu fabrication costs are less than $33/kg. Four different recycling schemes were considered; the best scheme was found to be blending the Pu with fresh natural U in standard Calder Hall fuel elements. This would be competitive with selling the Pu at the end of each run for $12/gm provided Calder Hall fuel elements containing up to 0.7% Pu can be fabricated for less than $33/kg and fuel element life is not limited by radiation damage. American Standard feels a realistic estimate for this cost is $20/kg.

The power density in a Calder-Hall type reactor is so low that over a century is needed to reach equilibrium Pu concentrations. The first six core cycles would cover a period of 72 years. For a $20/kg fabrication charge the average fuel cost with Pu recycle over this period would be 2.60 mills/kwh; if the Pu were sold at $12/gm the fuel cost would be 2.94 mills/kwh. This implies a value of Pu for Calder Hall reactors of about $24/gm.

Th-U^{233} Economics

Other work by American-Standard (*1044*) finds that thorium converters will achieve a fuel cost competitive with conventional fuel in the U. S. provided they can attain a fuel exposure of 20,000 Mwd/ton of Th and a conversion ratio better than 0.8. This conclusion resulted from detailed estimates for the present day and long range costs of the various fuel processing steps in the Th-U^{233} cycle.

Although the calculations were made for a hypothetical sodium-graphite converter (operating on thorium enriched with 2.3% U^{233} with a lifetime of 20,200 Mwd/ton), the conclusions would be valid for any converter with roughly the same lifetime and thermal efficiency. In the fuel cycle U^{235} makeup is added to compensate for conversion ratios less than 1.0. The authors estimate present day fuel processing costs at about $89/kg for ThO_2 (exclusive of fabrication costs); the long range value is put at $50/kg.

If the above reactor has a conversion ratio close to 1.0 it could afford to spend as much as $182/kg of Th for fuel fabrication on top of the present-day processing charge and still have a total fuel-cycle cost below 2.0 mills/kwh (the competitive level in the U. S.); the long range figure would be $220/kg. For a reactor with a conversion ratio of 0.8, the allowable present-day and long range fuel fabrication charges would be $113/kg and $152/kg respectively. Even for remote fabrication, these charges look not unreasonable today and should certainly be attainable eventually.

LMFR Pu Fueled

Brookhaven National Lab (*461*) suggests that the LMFR concept would have important advantages as a Pu burner and converter. The biggest of these would be the elimination of remote fuel fabrication. BNL points out that this is especially important for the steady-state Pu isotope mixture—Pu_∞, which would be one hundred times more radioactive than pure Pu^{239} (see box).

The Brookhaven Pu version of the LMFR (NU, April '56, 72) would be a single fluid reactor with graphite moderator and reflector but no blanket. The fuel solution would consist of 2,000 ppm-Pu_∞, 1,000 ppm natural or depleted U and Mg and Zr corrosion inhibitors dissolved in liquid Bi. A graphite-to-Bi volume ratio of 0.5 gives a core-plus-reflector critical diameter of 14 ft and a conversion ratio of 0.47. The reactor would yield 1.5 Mwd/gm of Pu^{239} feed. Exploratory work shows that a Bi solution of 20% enriched U would give about the same breeding ratio. In terms of this reactor design, Pu^{239} would have the same value as 20% enriched U or $16/gm.

Geneva 1958

CROSS SECTIONS

The 1958 Geneva cross sections data reflected more variety, more accuracy, and less mystery than the 1955 disclosures. Gone was the atmosphere of suddenly-revealed secrets; instead, this year's abundance of data permitted investigators to compare results from laboratories around the world, and to check their accuracy and agreement.*

Generally, the agreement was good and the accuracy improved over the 1955 figures. But there were still islands of disagreement—even on such basic parameters, for example, as the fission cross section of U^{235} and the absorption cross section of graphite.

In some areas, data was not so abundant. Resonance cross section information was scanty and agreement was poor for U^{235}, U^{233}, and Pu^{239}. Reactor designers found partial cross sections data on inelastic scattering and n-capture thin indeed.

Thermal Cross Sections

Although knowledge of thermal cross sections has always been more complete than for any other energy region, there is still a large amount of effort spent on cross sections which are important as standards, such as boron and gold, and on those important to reactors, such as those of the fissionable nuclides and structural materials.

Boron. The absorption cross section of normal boron (19.8% B^{10}) is essentially due to the B^{10} (n,α) reaction alone. Some uncertainty in the boron cross section arises from the variation in the isotopic abundance of B^{10} from sample to sample—a variation of ~1%. The value given in BNL 325† is 755 ± 4 b, and the value given in the second (1958) edition is 755 ± 2 b, expressed in terms of the "Argonne-Brookhaven" standard boron, which has a B^{10} content of 19.8 atom per cent.

Graphite. The 2200 meters/sec absorption cross section of graphite, essentially the cross section of carbon, has decreased slowly through the years as graphite samples improved in purity. Now, however, the process not only seems to have stopped, but to have reversed itself somewhat: The cross section quoted in the second edition of BNL 325, 3.4 ± 0.3 mb, is slightly higher than the 1955 value of 3.2 ± 0.2 mb. More recent results push the value still higher, so that it now appears that the absorption cross section should be about 3.7 mb. Further measurements and discussion will help to establish a definite value for this highly important cross section.

Fissionable nuclides. There were changes in the values for fission cross sections of U^{233} and Pu^{239} because of recent measurements, and a change in the U^{233} absorption cross section due to reevaluation of the U. K. value. There is still rather a wide spread in the recently-measured values of the fission cross section for U^{235}. One paper (*1599*) gives a direct measurement of 605 ± 6 b; this contrasts with Safford and Melkonian's indirect results which lead to a value of 581 ± 12 b. Finally, a Canadian paper (*204*) presents the value of 569 ± 6 b. Considered by themselves, the U. S. results are internally consistent, but the world results are not.

Fission products. For successful operation of high-flux reactors, which operate for long times before fuel removal, consideration of changes in reactivity resulting from formation and destruction of various nuclides is of great importance. Although the average cross section behavior of these nuclides in the resonance energy region can be roughly predicted from the distribution laws of level parameters (*1036*), the cross section in the important thermal region must be measured. It is impossible to predict the thermal cross section of a fission product, for example, because it depends on the chance proximity of a neutron resonance, whose location cannot be predicted with present theory. However, the situation is different for stable fission products, since samples are readily available and easily handled, naturally they are easier to work with than radioactive fission products, which are usually available only in small, highly active amounts.

The highly-important cross section of Xe^{135} has been measured as a function of neutron energy; so have a few others, (*673*) such as Sm^{151} and Pm^{147}, but for the bulk of fission products, thermal cross sections alone are known.

Resonance Parameters

Although cross sections of fissionable nuclides in the Mev region (where individual resonances are not observed) are now reasonably well known, our knowledge of resonances of the fissionable nuclides leaves much to be desired. Recent progress, however, indicates that in a few years the parameters of the fissionable nuclides will be reasonably well known.

The importance of this information to reactors rests primarily in the dependence of breeding gain and temperature stability on neutron interactions in the resonance region. The analysis of information on fissionable nuclides (for all but the lowest energies) is difficult not only because the fission width must be measured, but also because of the complexity of the resonance curves.

The values presented for the first few resonances in U^{233}, U^{235}, and Pu^{239} do not agree well; for higher energies there are still more gaps in the in-

Continued on p. 126

* This summary is condensed from U. N. Survey Paper *2483*, by Donald J. Hughes, Brookhaven National Laboratory, Upton, N. Y.

† D. J. Hughes, R. B. Schwartz, "Neutron Cross Sections," 2nd ed. (Brookhaven National Laboratory, Upton, New York, 1958).

Geneva 1958

MATERIALS

Effects of Radiation on Metals

Recent work at Harwell on the mechanical properties of irradiated metals and alloys included a wide variety of experiments on both single and polycrystalline metals (*80*). Hardening of copper single crystals by electrons was demonstrated, and the role played by interstitial atoms and vacancies was clarified by neutron and electron irradiation experiments at −195° C. Experiments on the effect of grain size on irradiation hardening in mild steel, and also the temperature dependence of irradiation hardening in polycrystalline copper, suggests that there are two basic irradiation hardening processes—locking of dislocation sources and increase in the resistance of the lattice to dislocation movement.

The experimental details and results are given in Table 1. In general, irradiation provides a marked increase in yield strength, a smaller increase in ultimate tensile strength and a moderate reduction in elongation to fracture. Several unusual features, however, were observed:

- The yield strength of both irradiated and unirradiated titanium was strongly dependent on the testing temperature, and the increase in yield strength produced by irradiation was about constant at 8,000 psi between −80° C and +200° C. At 200° C, however, a marked yield point was observed that was absent at 20° C.
- Very heavily cold-worked zirconium showed a slight reduction in yield strength on irradiation, but the rate of work hardening after the yield increased, and the ultimate strength of the irradiated material was slightly higher at all testing temperatures. The most striking difference, however, was that elongation before fracture of heavily cold-worked zirconium increased on irradiation whereas that of annealed titanium decreased.
- The yield strength of unirradiated molybdenum is strongly temperature dependent. Irradiation increased the yield strength throughout the temperature range but the increase was much greater at 200° C (15,000 psi) than at 20° C (5,700 psi). This is in complete contrast to the behavior of such metals as copper and nickel in which the yield strength after irradiation is much more temperature sensitive than before. Irradiated molybdenum also develops a marked yield point when tested at 20° C after annealing at 200° C for 60 min. This is attributed to the migration of vacancies to the dislocations locking them.
- In both copper and nickel, very large increases in yield strength were observed after irradiation. The hardening was very temperature sensitive, being much greater at low temperatures; e.g., at −195° C, the increases in copper and nickel were 284% and 193% respectively.

TABLE 1—Effect of 5×10^{19} n/cm² on Mechanical Properties

	Temp (°C)	Yield strength (10^3 psi)		Ultimate tensile (10^3 psi)		Elongation (%)	
		Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.
Mo	20	93.7	99.4	99.8	104.3	23.6	22
	90	80.3	93.3	90.5	93.5	23.8	18.4
	200	70.4	85.5	74.6	85.9	2.8	5.8
W	200	148	131	173	173	2.4	4.2
Ti	−195	—	—	151	165	—	—
	−78	101	109	107	112.1	10.6	4.8
	20	79.8	88.7	83.4	92.2	10.4	8.3
	200	37.2	46.2	53.1	55.2	9.4	8.4
Zr	20	107	106	124	124	3.5	4.2
	90	103	96	115	115	2.7	3.4
	160	92	90	101	106	1.9	2.9
	200	87	82	91	93	2.1	2.7
Cu	−195	10.7	41	46	53	40.7	21.6
	−78	11.2	34	34	40	38.4	21.2
	20	8.4	30.3	27.1	34	42.2	27.5
	200	7.9	23.1	22.5	24	30.9	15.9
Ni	−195	38	90.8	88	105	32	15
	−78	38	75	70	78	41	24
	20	36	62	59	63	34	23
	200	34	54	51	59	27	19

MTR Materials

At the time the MTR started operating in Idaho, in March 1952, radiation damage data on aluminum, stainless steel and beryllium were either not available or covered exposures equivalent to only one or two months service in the MTR (*1878*). Because of this fact, it was decided that coupons of metals of which the reactor core was constructed would be exposed continuously to as high a flux as possible.

It was planned to remove sets of these coupons after various exposures and test them to determine changes that had occurred in their mechanical properties. It was felt that the condition of these coupons at the time of testing would approximate the condi-

TABLE 2—Effect* of 10^{21}–10^{22} nvt on MTR Materials

Material	Neutron flux > 100 ev (nvt)	Rockwell hardness	Yield strength 0.2% offset (psi)	Ultimate tensile strength (psi)	Elongation in 1 in. (%)	Reduction in area (%)
1100 Al	0	6.3F	18,100	20,150	23.0	81.0
	0	10.4F	18,700	20,350	21.8	81.5
	2.4×10^{21}	36.5F	23,200	26,950	25.5	78.0
	7.2×10^{21}	46.9F	23,550	29,200	20.2	75.0
	2.6×10^{22}	42.3F	26,500	30,400	23.0	73.0
	3.76×10^{22}	47.2F	28,877	33,265	25.0	77.0
356 Al	0	29.6B	24,000	32,800	2.9	—
	0	29.0B	24,100	32,000	2.4	—
	2.4×10^{21}	60.6B	42,400	44,300	0.4	—
	7.2×10^{21}	64.3B	51,000	54,000	0.9	—
	2.6×10^{22}	46.6B	36,600	45,000	1.5	—
	3.76×10^{22}	58.4B	56,530	60,000	2.6	—
QMV Be	0	70.6B	23,200	35,000	1.4	—
	0	71.5B	25,800	36,400	1.4	—
	2.4×10^{21}	88,8B	—	51,400	0.2	—
	7.2×10^{21}	97.4B	—	50,400	0.2	—
	2.6×10^{22}	88.7B	>47,800†	—	—	—
	3.76×10^{22}	159.2B	†	—	—	—
347 SS	0	16.4C	66,300	101,800	57.0	74.5
	0	12.4C	56,900	89,800	69.4	81.5
	2.4×10^{21}	20.5C	98,000	109,000	51.6	78.5
	7.2×10^{21}	20.6C	97,300	109,700	50.6	77.8
	2.6×10^{22}	21.8C	100,800	107,250	44.0	65.0
	3.76×10^{22}	23.3C	106,939	115,306	32.0	74.0

* Measured at room temperature.
† Coupon broke in grips.

TABLE 3—Effect of Radiation on Strong Materials

Material	Neutron flux > 1 Mev (nvt)	Rockwell hardness	Ultimate tensile strength (psi)	Yield strength 0.2% offset (psi)	Elongation in 1 in. (%)	Reduction in area (%)
16-1 Croloy	0	83B	76,500	57,800	36	71
	4.9×10^{19}	96B	116,600	116,600	—	63
	7.8×10^{19}	95B	108,600	108,600	14	60
Hastelloy-X	0	87B	112,500	49,400	52	52
	5.0×10^{19}	96B	129,300	100,200	50	62
	1.8×10^{20}	99B	130,700	104,000	43	64
	2.5×10^{20}	100B	131,000	106,300	42	59
Inconel-702	0	82B	94,900	42,600	67	68
	5.0×10^{19}	96B	113,100	101,100	50	63
	1.8×10^{20}	100B	121,500	120,300	41	64
	2.9×10^{20}	101B	130,000	128,800	36	63
Inconel-X*	0	27C	170,100	103,200	28	36
	0.8×10^{19}	30C	165,100	143,300	22	30
	1.6×10^{20}	30C	161,300	158,600	14	43
Inconel-X†	0	32C	184,300	120,300	23	29
	0.8×10^{19}	34C	172,600	154,500	18	32
	1.6×10^{20}	34C	174,000	170,800	13	41
K-Monel	0	28C	163,400	132,400	22	45
	4.9×10^{19}	29C	168,700	167,600	12	42
	7.6×10^{19}	26C	167,600	166,500	11	43
410 SS	0	34C	177,000	147,700	20	66
	1.1×10^{19}	—	194,200	171,200	17	64
	1.0×10^{20}	—	201,000	182,800	15	64
	2.9×10^{20}	34C	204,700	188,900	14	58
	3.3×10^{20}	35C	204,400	189,300	14	57

* Double aged. † Singled aged.

tion of the reactor components constructed of similar materials.

The reactor tank, in the vicinity of the core, is constructed of 1100 aluminum. The lower grids supporting the reflector (QMV beryllium) and fuel elements, and the upper grid locating the fuel elements, are constructed of 356 aluminum. Stainless steel (AISI Type 347) is used for lifting pins, shafts, locating lugs and various positioning devices in core components. With the exception of beryllium, all the metals used in the reactor core were commercially available.

Coupons of these four metals were placed in a modified reflector piece next to the core of the reactor. This position is one of the highest thermal and fast flux regions in the reactor. The room-temperature mechanical properties of these alloys before reactor exposure and after the various exposures are summarized in Table 2.

Although changes are evident in the properties of the metals, these changes have not been great enough to date to impair the operation of any structural component.

Structural Materials

To determine the effects of reactor irradiation on the properties of commercially available structural materials that satisfy some of the requirements of high yield strength at operating temperatures, low coefficient of thermal expansion and high thermal conductivity, coupons of these materials have been exposed in MTR and ETR (*1878*).

The temperature of the samples during reactor exposures was 100° F and mechanical testing was performed at 75° F. All of these materials were in the annealed condition except that the K-Monel was age hardened, the 410 stainless was oil quenched from 1,800° F and drawn at 1,000° F and the Inconel samples were either single or double aged.

The results of testing the irradiated materials are given in Table 3. These materials have been subjected to comparatively very low neutron exposures. It will be noted, however, that the increase in yield strengths with reference to the increase in the ultimate strengths of some of the materials will probably prevent their use in highly loaded structures in a neutron environment.

Heavy-Water Production and Use

Savannah River

Heavy water is extracted from natural water at the U. S. AEC's Savannah River plant using a combination of three processes (*1065*):

1. Dual temperature exchange between hydrogen sulfide and water. The product from this step contains about 15% D_2O. The ratio of separation factors at the two different temperatures is a measure of the effective separation factor, which in practice is usually in the range of 1.1–1.4.

2. Vacuum distillation (hereafter called the DW process), which operates with a separation factor of 1.05 at 100 mm Hg. The product from this step contains about 90% D_2O.

3. Electrolysis (hereafter called the E process), which has a separation factor of about 8. The product from this step contains 99.75% D_2O, the desired end product.

Isotope separation is usually accomplished by means of a cascade, a large number of separating steps connected

in series. With a dual-temperature exchange cascade that comprises successively smaller towers, more than 90% of the separation, and approximately the same fraction of the cost, is expended in reaching a concentration of 1%. Hence, the problem of obtaining heavy water is essentially solved, as regards engineering economics, once the concentration reaches 1 or 2%. Beyond this point both the equipment and energy required will be small.

The value of the product handled in the second process, however, will be correspondingly large, and, for this reason, it is important that losses be kept to a minimum, and, thus, first consideration should be given to those that are simplest to operate and entail the least risk of product losses, rather than to those that require the least energy per unit of separation.

The energy consumption of a DW-process cascade is high compared to some other methods of separation, but this is not significant. (This may be illustrated by noting that the flow in a cascade, at the step where concentration has reached 15%, is down by a factor of 1,000 compared to the flow at the feed point of the cascade.)

Energy required by the E process is also high per unit of separation. This is not significant when handling high concentrations of D_2O, but does become worth noting if the process is used to enrich material containing 1 or 2% D_2O, or even 10% D_2O. (Savannah River confines the use of electrolysis to materials containing at least 70% D_2O.)

Economics of production. The Savannah River heavy water plant, built in 1951–52, cost $164,000,000 (*1065*). This includes the cost of facilities supplying steam and electric power for process use. Of this total, about $2,500,000 represents the cost of the distillation plant and about $1,500,000 the cost of the electrolytic plant.

The maximum production rate is about 1,000,000 lb D_2O/yr. Hence, the "investment per annual pound" is $164.

The operating cost, exclusive of capital charges, is about $14/lb D_2O, when producing at full rate. An analysis of the operating cost is shown in Table 4. Interestingly, the U. S. sells heavy water at a price of $28/lb—very approximately the Savannah River cost.

Heavy-water recovery. Each production reactor at Savannah River is equipped with two vacuum distillation towers that operate continuously to remove light water from the reactor system (*1065*). The two towers operate in series. The feed is moderator from the reactor system, circulated from the reactor through the base of one of the towers and back to the reactor. The "overhead product" from the second tower contains about 90% D_2O. This material is sent to the DW plant for upgrading.

An entry rate of 6–8 lb/day of light water is perhaps a reasonable figure for a plutonium-production reactor (*610*). The cost of removing the light water can be estimated from the approximation that the 1,000 lb/hr of steam used to heat the calandria of a well-designed vacuum distilling tower will remove 1 lb H_2O/day from a 99.75%-D_2O liquid as a 75%-D-25%-H mixture.

It is not absolutely required that the moderator be maintained at this purity. If a purity of only 99.5% deuterium is maintained, the size of the purification equipment and the amount of steam used can be reduced by a factor of two. As an alternative, the entry rate of light water can be allowed to increase by this same factor. The loss in reactivity that results from this extra light-water content may completely overbalance the savings in isotopic purification equipment. The loss in reactivity is 1.5–3% for each mole% increase in H_2O.

TABLE 4—D_2O Production Costs

	Savannah River ($/lb D_2O)	*Nangal, India ($/lb D_2O)*
Feed water	0.40	—
Hydrogen loss	—	2.92
H_2S	0.44	—
Inert gas	—	0.39
Wages	2.62	1.62
Electric power	1.33	4.86
Steam	4.40	—
Cooling water	0.12	—
Maintenance (materials)	0.49	1.78
Misc. supplies	0.21	0.49
Overhead	3.49	0.81
Total operating	13.50	12.87
Capital charges	16.4*	14.97
Total cost of D_2O	29.9	27.84

* 10% of "investment per annual pound" given in text.

The losses of D_2O are another item of concern. The experience at Savannah River is that losses of D_2O during the past two years have averaged about 3%/yr. The lowest rate of loss from any single reactor for a six-month period was 2%/yr. These figures are based on the total inventory of heavy water in the reactor and such associated auxiliaries as pumps, piping, etc.

India

A plant to produce heavy water is under construction in India, on the banks of the Sutlej river, at Nangal, in the Punjab (*1649*). This plant is expected to produce 14 metric tons of heavy water per year and 70,000 metric tons of nitrogen as nitro-limestone.

Investigations of the feasibility of producing heavy water in India began early in 1954. At that time, information was available on only two processes—electrolysis and water distillation. Because it was felt that water distillation was a very expensive process, electrolysis of water was adopted as the method of primary enrichment with another, then undetermined, method for final concentration.

In response to an invitation, detailed project reports were received from three firms in August, 1956. Although all the processes suggested took advantage of the concentration obtained in the electrolyzers, different processes for the final D_2O enrichment were recommended, namely, ammonia distillation, catalytic exchange followed by water distillation, and hydrogen distillation.

On the basis of the data supplied by these firms in their reports, technical and economic evaluation of the three processes were made. As a result of this study, it was concluded that although the catalytic exchange process was a well proven process and had a background of 10 years of industrial scale operation, the hydrogen distillation process treating a reduced amount of enriched hydrogen would be economically more attractive and technically more suitable for conditions in India. A cost analysis made by the Indian Government for a typical H_2 distillation plant is given in Table 4.

For more on MATERIALS see p. 112.

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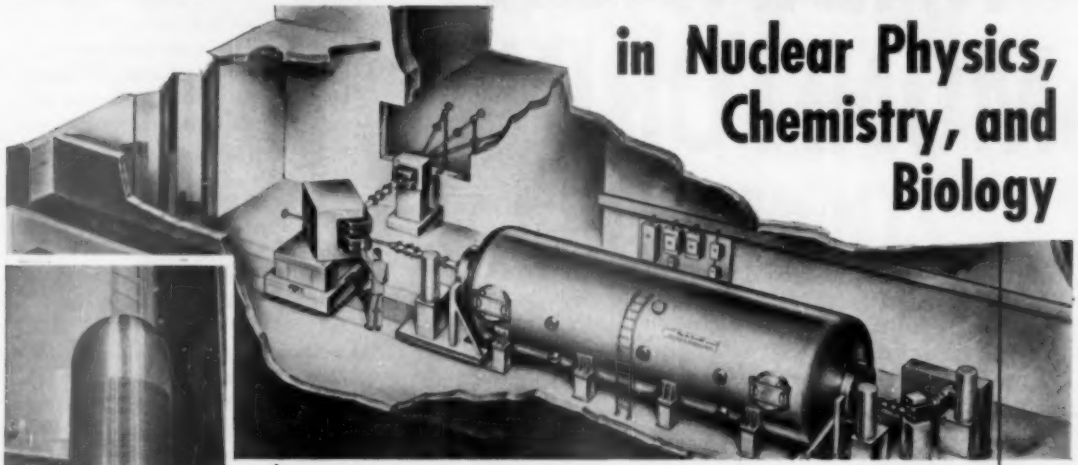

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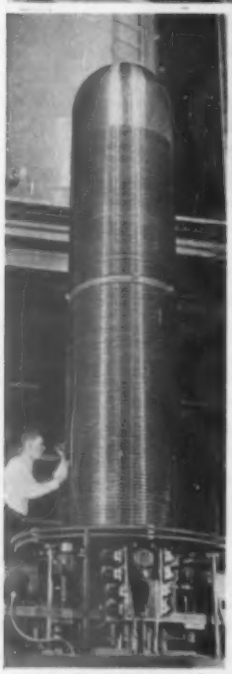

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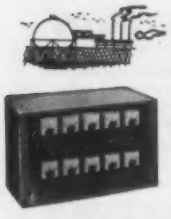

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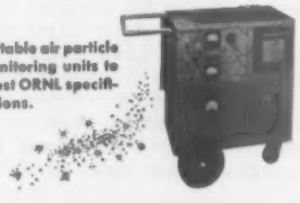

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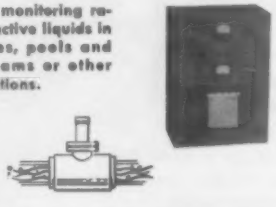

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MATERIALS

This article starts on page 108

Zirconium

Hydriding Attack

In-pile-test failures of Zircaloy cladding on fuel elements have been accompanied by a redistribution of dissolved hydrogen to the outer surface of the cladding with occasional formation of a rim of zirconium hydride on that surface (*709*). The existence of a thermal gradient of about 525° C/cm between the inner and outer surfaces of the cladding led to the belief that the hydrogen redistribution was caused by thermal diffusion of hydrogen from the hot to the cold side of the Zircaloy cladding.

Work was undertaken in the U. S. to observe the nature of the thermal-diffusion phenomenon in zirconium-hydrogen solid solutions and to measure the parameters characteristic of the process. From the experimental results, the extent of hydrogen redistribution in core under operating conditions has been estimated. These show that, if the critical hydrogen concentration for a Zircaloy plate in a given temperature gradient and at a fixed surface temperature is exceeded, a Zr_2H_3 layer will form on the cold side. The thicknesses of the hydride rims formed for various initial conditions were calculated and plotted in Fig. 1. It can be seen that a greater thickness of hydride layer forms as the thermal gradient increases and as the magnitude of the heat of transport increases.

An interesting effect of β heat treatment of wrought Zircaloy-2 tube in reducing the severity of hydriding attack is illustrated by the results of Canadian work listed below (*194*). The weight gain (in mg/dm²) was:

	At 480° C		*At 540° C*	
	20 hr	*48 hr*	*10 hr*	*48 hr*
As drawn	63	2,530	1,695	3,860‡
α-annealed*	53	2,550	2,550	3,180‡
β-annealed†	56	65	87	170‡

Development of Alloys

British investigations give slim hope for outstanding improvements in the corrosion resistance of zirconium in gaseous media by alloying; improvements greater than a factor of two will not be forthcoming at any temperature, although considerable improve-

* Vacuum annealed at 700° C for 1 hr.
† Vacuum heat treated for 3 hr at 980° C and air-cooled in sealed tube.
‡ Including weight of loose oxide.

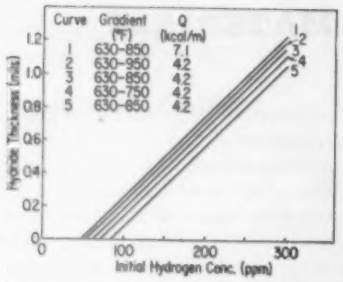

FIG. 1. Thickness of Zr_2H_3 precipitate at steady state as a function of initial hydrogen concentration for various temperature gradients and heats of transport (Q)

ments in properties of film adherence in the post-transition period have been achieved (*1450*). In view of this, and despite any substantial improvements in creep resistance which might be obtained with Zr-Mo or Zr-W alloys, the use of zirconium in CO_2-cooled reactors at temperatures in excess of 500° C is not visualized.

Zirconium alloys have been developed that are more resistant to radiation-induced corrosion in uranyl sulphate solution, say metallurgists in the U. S. (*1993*). This has been accomplished by alloying Zr with Nb, Pd or Pt. The Zr-Nb base alloys can be heat treated to attain high strengths with moderate ductilities, but, under many conditions, brittle alloys result.

Zirconium Production

A West-German process separates hafnium by distribution of the complex zirconium and hafnium thiocyanates between two liquid phases (*1003*). The Zr goes primarily into the aqueous lower phase (containing the ions SCN′, H′ and Cl′), and Hf preferentially enters into the upper phase of methyl-isobutyl-ketone containing thiocyanic acid.

Reduction of the $ZrCl_4$ to metallic Zr is carried out essentially by the Kroll process, but changes have been made with a view to cutting down the time required for the reduction cycle plus distillation. In the usual Kroll process, reduction and distillation are performed in two different furnaces. This entails loss of time as a result of unavoidable cooling down and reheating, and the quality of the sponge is impaired by the increase in oxygen and nitrogen content when transferring to the distilling furnace. A furnace was devised, therefore, that is equally suited to both reduction and distillation. With this furnace, both processes can be carried out in succession

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Separation of Hf by selective reduction of $ZrCl_4$ with aluminum is being tested in India (*1672*). Starting with the mixed tetrachloride vapor, a partial reduction with Al is carried out so that the vapor phase gets progressively enriched in hafnium while the nonvolatile zirconium reduction product becomes substantially free from the impurity. The relative instability of the lower oxidation states of hafnium contributes to an improved separation factor.

This process provides an effective means for removing hafnium. It admits, however, of many improvements in technique. At present, two treatments are necessary to achieve a reactor-grade product, although high elimination of Hf occurs in the first treatment. It is expected that the efficiency would be much improved with a more rigorous control of temperature and with provision for disturbing the crust as it forms.

The inert-gas-shielded, molten-salt, electrorefining technique has proved adaptable to improving the quality of a number of metals (*698*). But, because of mechanical difficulties, it has not been possible, to date, to obtain sustained Zr production in the U. S. using chloride electrolytes (NaCl-ZrCl and KCl-$TiCl_7$-ZrCl). The fluoride electrolyte (NaCl- or KCl-K_2ZrFl_6) offered easier operation but apparently did not reject impurities as efficiently as the chloride. This assumption is based on data that show impurity content increasing in consecutive deposits, contrasted with the usual progressive improvement in quality from chloride electrolytes. The simple binary mixtures of NaCl and ZrCl have been difficult, if not impossible, to stabilize. A number of operational difficulties were encountered when zirconium chloride was used because of bath instability. Zirconium chloride subliming from the bath caused changes in bath composition, sticking of valves and plugging of orifices.

In Russia, a method of obtaining ductile zirconium metal by the electrolysis of fused salts has proved more successful (*2047*). The quality of zirconium produced by the electrolysis of the fused KCl-K_2ZrF_6 mixture was determined by spectrochemical analyses. Typical results show a total impurity content of $<1.2\%$.

Graphite

Nuclear Annealing

Early in the study of radiation damage to graphite, it was observed that graphite that had been damaged by irradiation could be annealed by a second irradiation at a higher temperature (*614*). The temperature of this second irradiation was considerably less than the temperature required to do an equivalent amount of purely thermal annealing.

The total interlayer-spacing change demonstrates that irradiation enhances annealing. It is revealing, however, to study the manner in which interlayer spacing damage is distributed with activation energy in samples that have been subjected to nuclear annealing. From a determination of damage distribution curves, it was concluded that kinetic processes with a wide range of activation energies contribute to nuclear annealing at any instant. Irradition at 335° C is effective in partially removing damage out to high-activation-energy regions reached by thermal annealing only at 1,300° C.

The mechanism by which nuclear annealing occurs has received only very slight attention, but it appears to be closely associated with the mechanism of contraction during high-temperature irradiations. Activation energies in excess of those available from equilibrium lattice temperatures are necessary for both. It is likely that nuclear annealing is accomplished through collisional processes that transfer energy in the range of 3–25 ev to the lattice defects, thus supplying the activation energy for periods of time long enough for diffusion to occur.

The routine annealing operation for the Brookhaven National Laboratory reactor is planned so that fission heat "triggers" the release of low-temperature stored energy (*462*). Fission heat is added continuously while the graphite temperatures are gradually elevated through the triggering region (80–150° C). The rate at which fission heat is removed from the reactor by means of the fans (usually 1–2 Mw) is high compared with the expected rate of propagation of the stored-energy release. If the fission heat removed from reactor structure while raising graphite temperatures to 150° C is large compared with the energy stored in the graphite, the release of the stored energy has little effect on structure temperature. Removing fission heat at a

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MATERIALS

This article starts on page 108

0.5-Mw reactor power level was found to be sufficient to mask the graphite-energy release.

The routine annealing procedure is performed at the end of a normal two-week operating period by shutting down the reactor according to a special procedure. The procedure commences with the reactor in equilibrium full-power operation. Power is reduced gradually over a period of 2 hr, and, at the same time, cooling is decreased in such a manner as to maintain the fuel temperatures constant at the maximum of 350° C.

During the power-reduction period, the graphite temperatures in the cooler regions gradually rise as cooling is decreased. When the air flow is reduced to approximately 10% of full value, and the power level is 1–2 Mw, the coolest graphite temperatures will normally have risen to above 150° C. The cooling load is then further decreased, and, finally, all fans are stopped. By continuing to operate at approximately 0.5 Mw and completely covering the inlet ducts with light sheets of plastic, the central graphite temperatures are driven up to 310–340° C. Finally, to complete the operation, the reactor is scrammed and allowed to stand for 2–3 hours with no air flow. Prior to taking measurements, the structure is cooled at a rate of 2° C/min by readjusting the cooling load.

Graphite Protection

Attempts to improve the resistance of graphite to in-pile oxidation have been sought in Britain along three lines: reduction of the accessible surface area by impregnation with carbonaceous or other materials, rendering the accessible surface area inactive and application to the visible surface of an impermeable, nonreactive coating (*305*).

Reduction of accessible surface area. Specimens densified by pitch impregnation, and subsequently regraphitized, led to a decrease in accessible surface area by a factor of about two. The results of tests for the impregnated specimens imply that the concept of "active sites" is of little significance in-pile, the irradiated gas apparently being sufficiently active chemically to react with any carbon atom irrespective of environment. The rate-determining factor is evidently the production of some form of chemically active species in the gas. These re-

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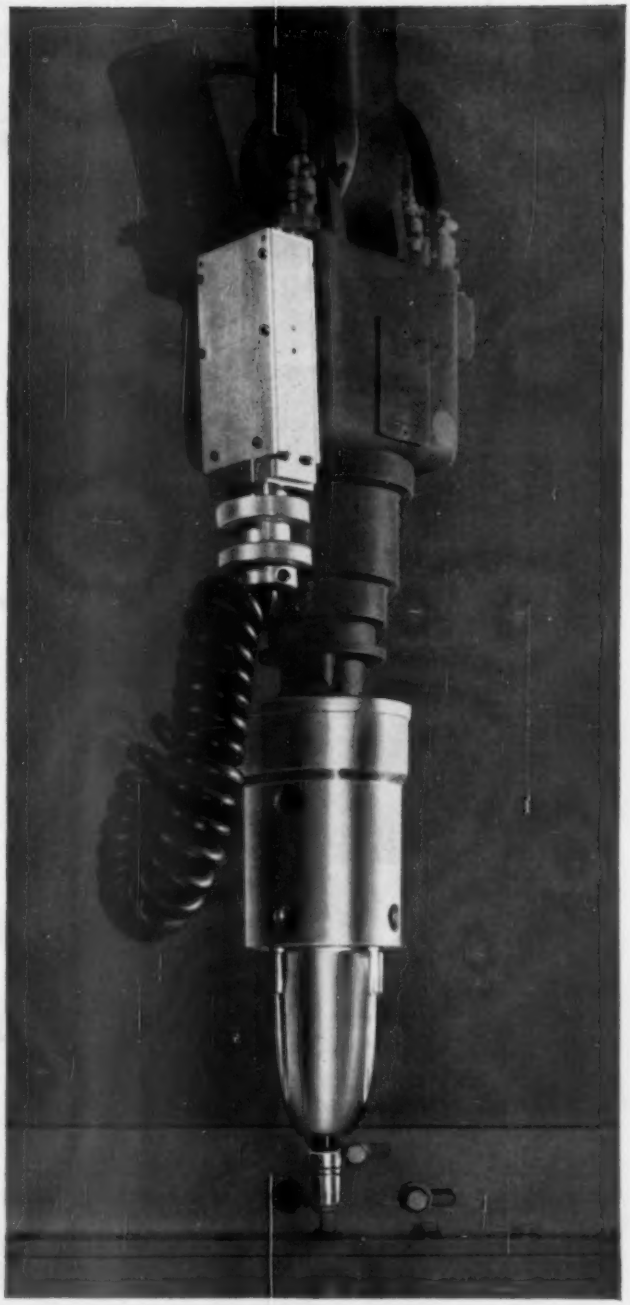

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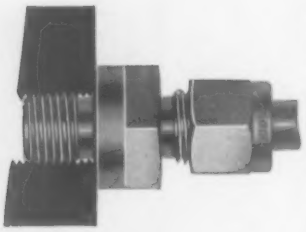

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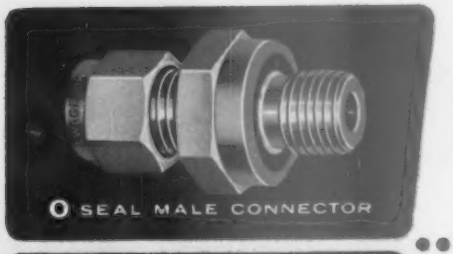

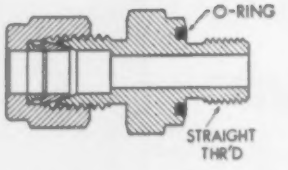

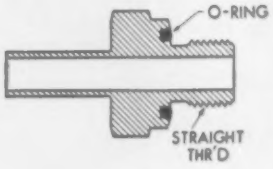

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sults possibly imply that oxidation occurs to a limited depth under these conditions.

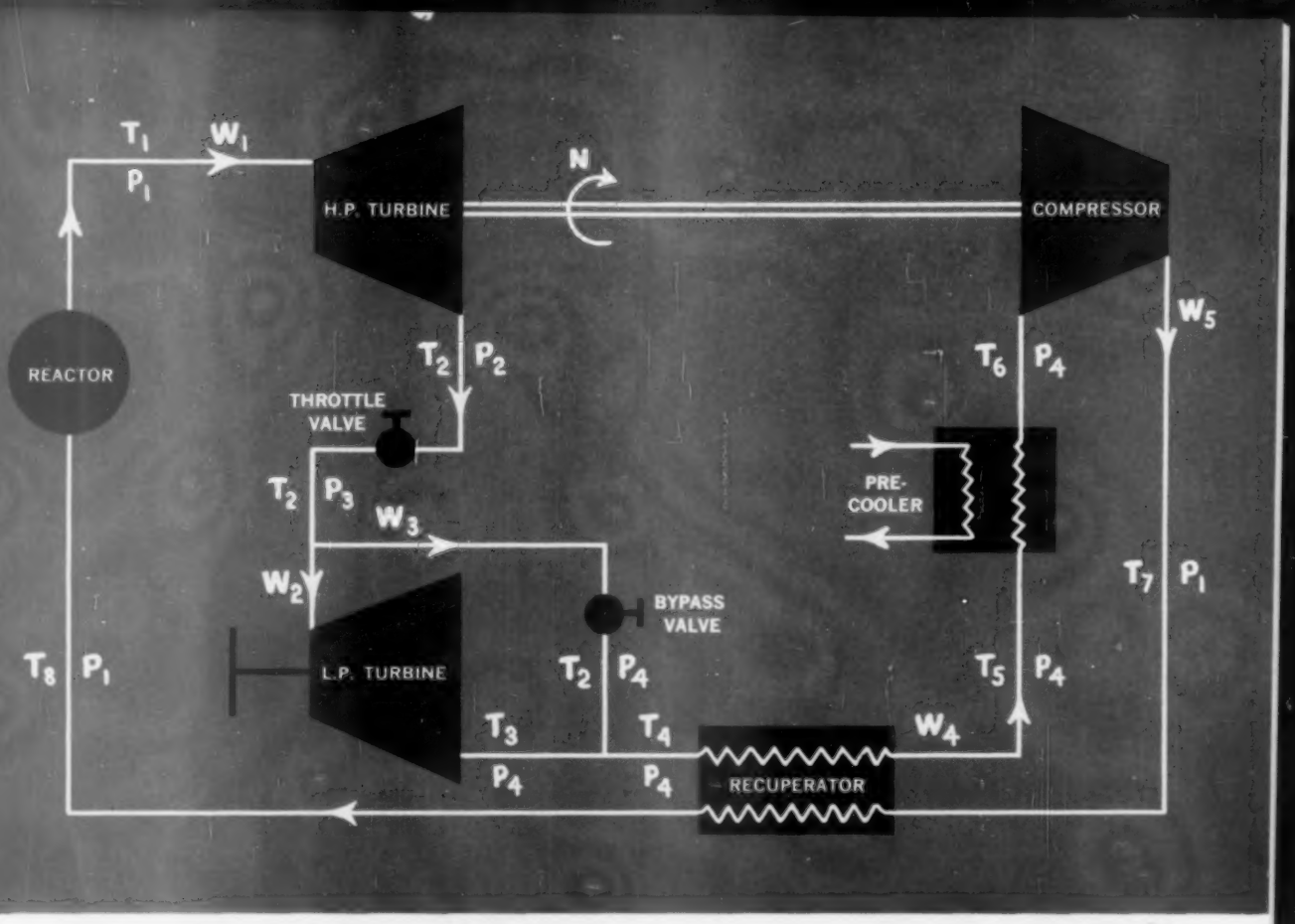
Rendering surface inactive. This type of treatment aims not to alter the accessible surface area but to render it less oxidizable. One method of achieving this is the addition of small quantities of phosphorus compounds to the graphite. A suitable treatment consists, for example, of exposure of the graphite to the vapor of refluxing trimethyl phosphate followed by rapid heating to about 1,000° C in argon.

The quantity of phosphorus in the graphite is of the order of that required for the chemisorption of a monolayer of a species such as P_4O_{10} or the coverage of edge sites with phosphate groups. In this way, quite small quantities are effective in blocking the active sites from which gasification normally occurs. With the phosphate-treated specimens, either the accessible surface area is not fully covered, and the uncovered portion, though inert out-of-pile, is oxidizable by the irradiated gas, or, for steric reasons, the treatment is ineffective in preventing access of an active species to the graphite.

Application of coatings. The third treatment included gas-phase deposition of a layer of silicon carbide. Specimens subjected to this treatment were given a prior impregnation and regraphitized to provide an improved substrate. Coating was used at a thickness of a few thousandths of an inch. The implications of the results for the silicon carbide coated specimens are less clear-cut. The similarity of oxidation rates for the coated specimens and the controls in the in-pile tests suggests that the underlying graphite is being attacked rather than the silicon carbide.

Several methods have been considered for coating graphite with NbC, TaC, and ZrC in the U. S. (*1428*). Of these, vapor deposition offered the most promise of forming continuous, adherent coatings. However, with the vapor-deposition procedures, the uniformity of the coating depends strongly on the vapor-flow pattern. Control of the vapor-flow pattern is practical in coating some shapes, but very little control can be exercised in coating small-bore tubing.

To overcome the problem of nonuniform reaction, workers at Battelle Memorial Institute have developed a



Digital analysis tests reactor system

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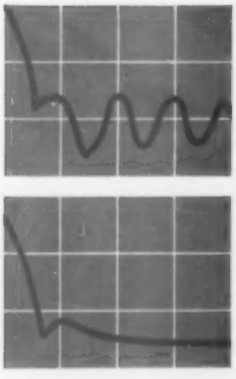

The problem was to evaluate performance of the complete reactor power system design, shown blocked out in preliminary form above.

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Partial results of digital analysis are shown in the two available power vs. time curves at left. They show the effects of a change in throttle and bypass valve settings. Initial analysis revealed the poorly damped oscillatory curve at top left. After a simple system modification, reanalysis produced the smooth change and quick settling of the lower curve. *All without a nickel spent on hardware!*

Reactor system simulation is only one of the nucleonic activities at Ford Instrument Co. Others include: Design of a complete closed-cycle gas-cooled reactor; development of fuel elements, including high temperature studies; manufacture of control rod drive mechanisms (e.g., those used in the submarine *Seawolf*); control rod position indicators; magnetic and transistor amplifiers for nuclear applications; and studies of control rod and reactor materials. In addition, Ford Instrument is conducting numerous feasibility and experimental studies in the nuclear field, including the application of digital techniques to reactor control.

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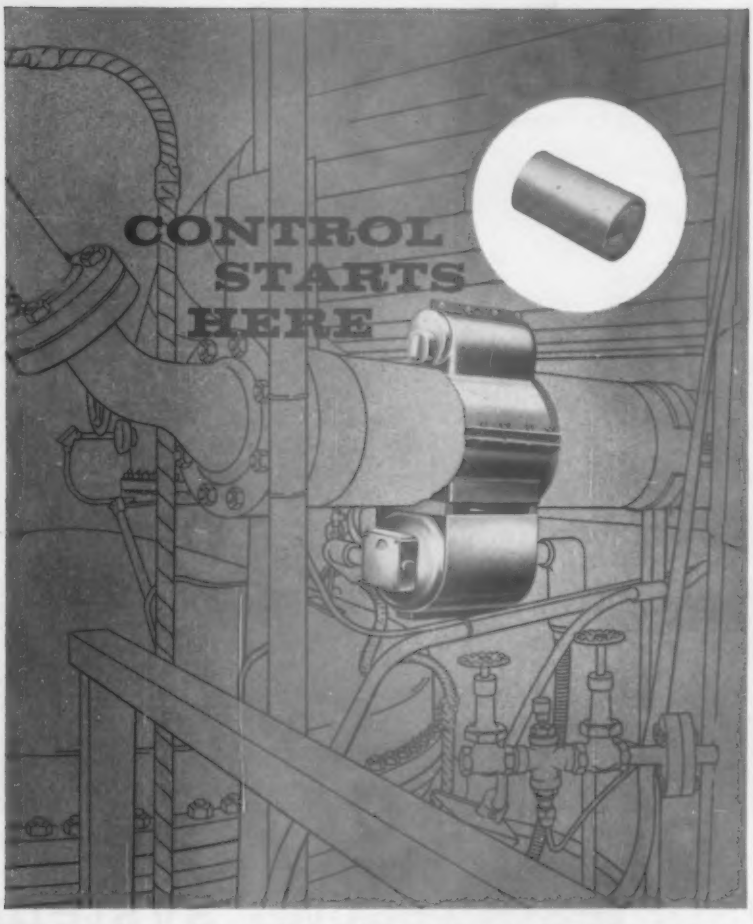

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vapor-deposition method in which the rate of coating formation is controlled by the rate of diffusion of carbon through the carbide. Under these conditions, the rate of coating formation is practically independent of the vapor-flow pattern, and a uniform coating results.

Present research has demonstrated the feasibility of uniformly coating extended graphite surfaces with metal carbides by reaction with metal halide vapors in a range of pressure and temperature unfavorable for the deposition of metal but favorable for carbide formation. In this range, the rate of diffusion of carbon through the carbide can be made to limit the coating rate, and uniform coatings are obtained over a wide range of vapor-flow conditions.

Sodium

Compatibility of reactor materials in flowing sodium has been studied in Britain (*25*). The materials studied fall naturally into three classes:

(a) Those whose oxides or complex oxides are of less than or comparable stability to that of sodium oxide. These include all the steels and nickel;

(b) Those possessing oxides or complex oxides more stable than sodium oxides and whose corrosion product is adherent to the parent metal (e.g. Zr);

(c) Same as for (b) but the corrosion product is nonadherent (e.g. Nb, V, U, Ti, Be).

Class (a) can be safely used in sodium circuits under cold-trapped conditions. Although work has shown that, in class (b), Zr forms a protective oxide, cold trapping is expected to be inadequate because of the deterioration of mechanical properties of the metal caused by oxygen embrittlement. It is essential that the sodium be thoroughly deoxidized for the metals of class (c) to be compatible with the liquid metal. Radioactive mass transfer for stainless steel and zirconium is extremely low, but activity arising from fission-product transfer from uranium in contact with sodium can be considerable, especially under cold-trapped conditions.

The inherent inertness of sodium with respect to graphite, zirconium and austenitic stainless steel may be negated by sodium-borne impurities, summarizes U. S. workers (*705*). Oxygen transported by Na can form a fatigue-sensitive surface layer of ZrO_2 on Zr

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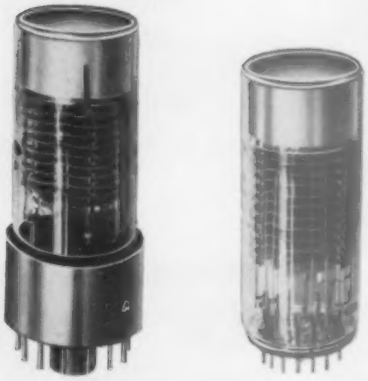

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MATERIALS

This article starts on page 108

core components. This diffusion-controlled process is slowed by the reduction of oxygen content in sodium to levels well below 10 ppm. Stainless steel is unaffected by oxygen content below 50 ppm sodium. Hydrogen released from the moderator graphite at high temperature diffuses slowly through the surface oxide on zirconium to form a solid solution. Rapid cooling of hydrogen-containing metal can cause delta-phase zirconium hydride to precipitate at zirconium grain boundaries, but this will dissolve again upon slow heating. Carbon in sodium can result in appreciable carburization of austenitic stainless steels at temperatures above 550° C with consequent deterioration of high-temperature fatigue properties. The control of oxygen content to the level of 10 ppm required for satisfactory operation in a system containing Zr can be achieved by hot-trapping a side stream of sodium with thin Zr or Zr-Ti alloy sheet maintained at around 650° C.

In sodium or NaK circuits where oxide level is controlled by cold trapping alone, the corrosion of beryllium specimens in the circuit is excessive, concludes Australian engineers (*1091*). The mechanism of attack is reduction of sodium oxide by the beryllium. The rate-controlling factor appears to be diffusion of oxygen ions, but the inertia of the flowing sodium has an additional effect similar to erosion. Protection of beryllium by preferential oxidation of thorium in a hot trap does not appear promising. The rates of weight gain of beryllium in cold-trapped sodium are much greater than those of thorium for the same temperature. Thorium oxide formed is adherent and largely protective, thus reducing hot-trapping efficiency, but the beryllium oxide formed is nonadherent and nonprotective. Acceptably low corrosion of beryllium can be attained, however, in dynamic sodium and NaK systems by suitable cold trapping followed by calcium deoxidation of the liquid metal. No mass transfer of the calcium or reaction with the nitrogen cover gas occurs, and reaction between calcium and nickel in the stainless steel remains at an acceptably low level.

Corrosive attack of stainless steels by sodium is accounted for by oxygen and carbon impurities in the sodium and also by oxide impurities in steels, say Russian scientists (*2194*). Other conditions being equal, steels that are well deoxygenated and melted in vacuum should have a strong resistance to corrosion. Steels and alloys containing a larger amount of noncarbide-forming element (such as nickel) have a lesser tendency towards carburization.

Polyphenyls

The advantages of polyphenyls—low vapor pressure, low induced radioactivity, lack of corrosive action on materials of construction and relatively good stability towards irradiation—have been confirmed in experience gained at Harwell (*51*). In particular, the slower rate of destruction when the coolant contains a large proportion of polymerized material has been reaffirmed. Further experiment is still needed to determine how much polymer can be tolerated.

The salient new fact arising from these experiments is the much greater destructive effect of energy derived from fast neutrons than of that deposited by gammas. The relative contributions of these two sources of ionizing energy will vary with position in the core, so that the absolute energy deposited in the coolant can't be estimated with precision.

The calculated damage yields for fast neutrons are so much larger than the electron yields that it seems inevitable that the contribution to power cost of maintaining a constant polymer content by continuous or periodic replacement with fresh coolant will prove to be appreciably greater than current estimates. This would have two consequences:

The first would be an increased incentive to work at the highest polymer content consistent with satisfactory engineering properties of the fluid to minimize its rate of breakdown.

The second consequence would be the necessity of using the cheaper polyphenyls, such as biphenyl and Santowax-R. These are the primary products of current production methods. The individual terphenyl isomers, which must be obtained by fractionation of the Santowax, cost more and are in relatively short supply. The use of fluids compounded from any of these would become still less attractive.

Experiments at Hanford have shown that the radiolytic decomposition rates of all polyphenyl compounds are similar (*611*). The main difference in stability lies in their resistance to pyrolytic degradation. From a practical

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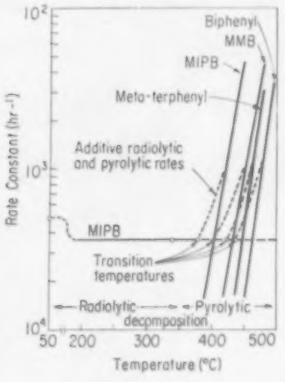
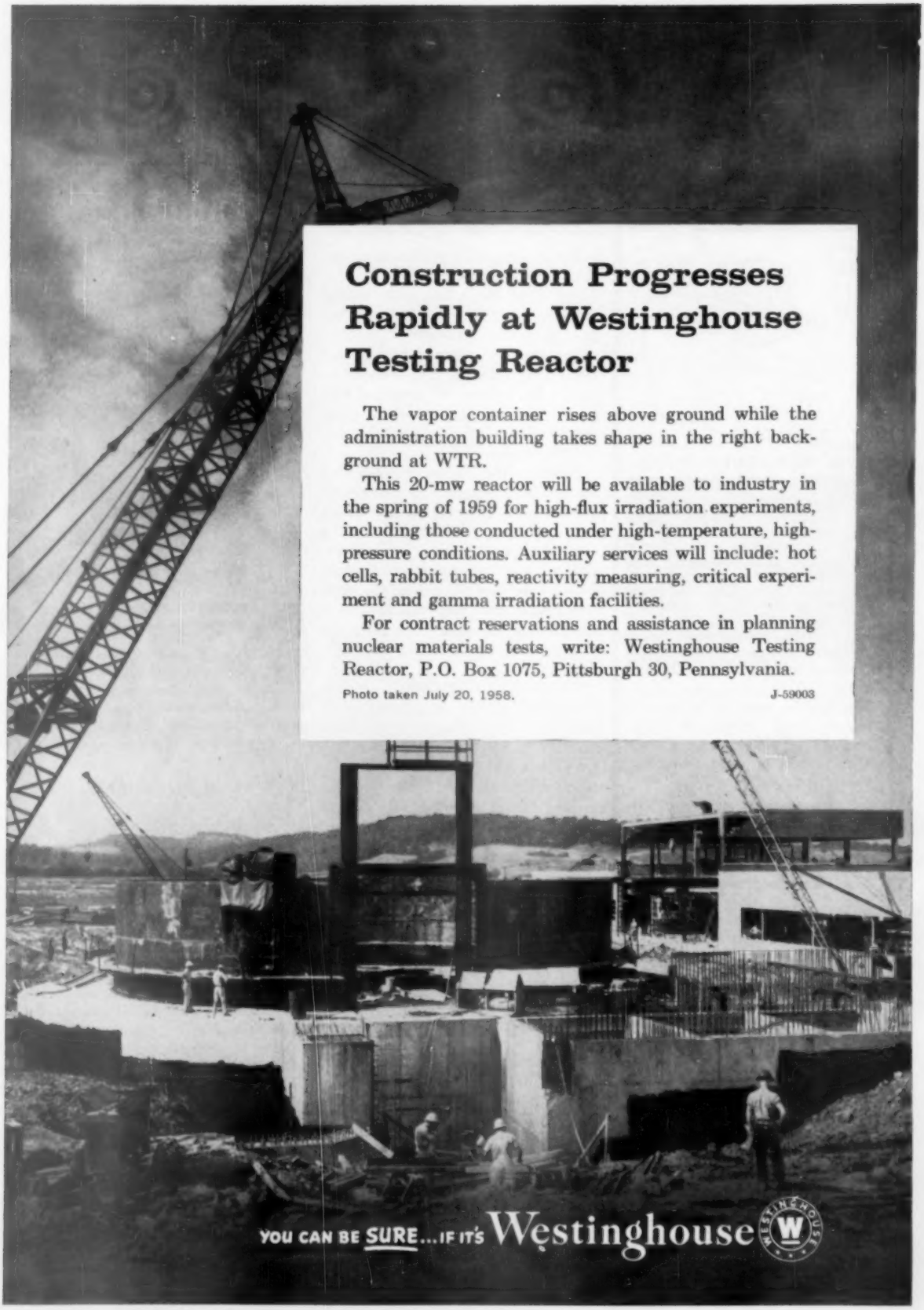


FIG. 2. Transition temperatures for organic coolants. Transition temperature is when pyrolytic decomposition begins to significantly increase over-all decomposition rate

standpoint it is desirable to operate the coolant at a temperature below the transition temperature where the pyrolytic decomposition begins significantly to increase the over-all decomposition rate of the coolant.

Radiolytic and pyrolytic data have been combined in Fig. 2. Since the radiolytic rates of the polyphenyls are all quite similar, only MIPB (monoisopropylbiphenyl) is used for the radiolytic portion of the graph because these data demonstrate the effect of irradiating the coolant at intermediate temperatures. Note that there is only a slight difference between the rates at 50° C and 340° C, indicating that the only effect of temperature on radiolytic decomposition is to slightly decrease the amount of polymer formed.

In a reactor that is constructed so that the radiolytic decomposition rate is higher than that chosen as an illustration in Fig. 2, the transition temperatures would be somewhat increased, and, hence, it would appear that the coolant could be utilized at a higher temperature. This is a consequence of assuming that it would not be desirable to operate a reactor at a temperature so high that the decomposition rate would increase above that caused by radiation. If the acceptable decomposition rate is above that resulting from radiolytic decomposition, the coolant temperature can be correspondingly higher. Based on Fig. 2, the approximate transition temperatures are: biphenyl—435° C, terphenyl—425° C, MMB (monomethylbiphenyl)—400° C and MIPB—375° C.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CROSS SECTIONS

This article starts on page 107

formation and the agreement of results is poorer still. Agreement is bad for radiation widths and worse for fission widths; more experimental and theoretical work is certainly necessary to clarify the situation.

Partial Cross Sections

Determination of partial cross sections requires much higher neutron fluxes than do total cross section measurements, and further difficulties arise from the necessity of absolute measurement of neutron flux or neutron energy. Partial cross sections are still poorly represented compared with total cross sections, but definite progress has been made in the last 3 years. Some examples:

Fission. The fission results show marked improvement, both in resolving power in the ev region and the absolute values in the Mev region. For U^{233}, in contrast to the 1955 information, we now have a number of results from laboratories throughout the world which agree extremely well in spite of the difficult questions of counter efficiency and absolute flux measurement involved.

The results for U^{238} show similar improvement. There is definite structure in the fission cross section at about 0.7 Mev, as there is in the capture n, γ cross section. The onset of the $n, 2n$ reaction is reflected in the fission cross section curve as well. The definite step in the fission cross section at 1.6 Mev occurs in other nuclides at nearby energies. One paper (*2467*) discusses these features in terms of nuclear models.

Capture. Despite the reactor designer's need for information on capture cross sections, data has accumulated very slowly. Inspection of the second edition of BNL 325 reveals a definite gain in the number of cross sections measured, but unfortunately the results here do not agree as well as the results on fission cross sections. The situation is well illustrated by the absorption cross section of gold, for which there are a number of measurements, in very poor agreement.

Nonelastic and inelastic cross sections. Results have been available for some time for a few nuclides, but these cross sections are particularly difficult to measure because of the necessity to determine the energy distribution of the scattered neutrons for each value

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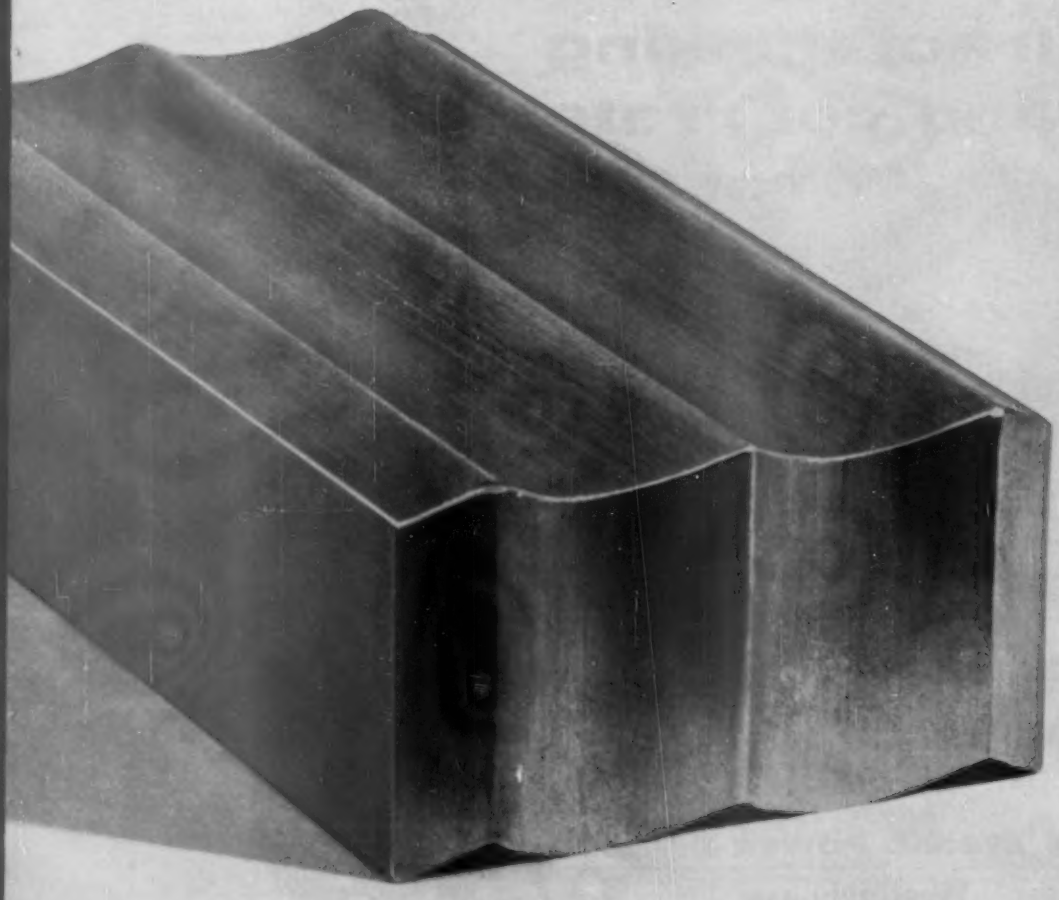

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BETTER YOUR FUTURE AT BETTIS

This article starts on page 107

of the incident energy. Recently techniques of time-of-flight measurement of scattered neutrons have shown promise, although very little data is yet available. One method based on measurement of the gamma rays associated with inelastic scattering has given information on about 20 nuclides; for neutron energies not far above the inelastic scattering threshold the gamma spectrum is simple and gives complete information on inelastic scattering. However, there seem to be some serious difficulties in this method.

The nonelastic cross section represents all partial cross sections except elastic scattering and is relatively easy to measure by sphere techniques using threshold detectors. In general, the nonelastic cross section can include many processes other than inelastic scattering, but in practice (and for energies not too high above the inelastic scattering threshold) these other processes are not significant. Thus for many cases the nonelastic cross section is essentially a measurement of the inelastic scattering.

Prediction from Theory

Reactor design requires prediction of a large number of neutron cross sections over a wide energy range. In the past, thermal cross sections have been the primary data used in design and these are sufficient for rough estimates, particularly for thermal reactors. However, at the present time not only is it desirable to design reactors that operate at temperatures other than thermal, but also to consider the effects of resonance neutrons even in thermal reactors themselves. For reactors operating at high power, it is essential that cross sections of isotopes built up during operation be considered.

The combination of measurements made at low energy with theories based on nuclear models could be of great value in reactor calculations, for this would allow calculation of average cross sections, self-protection effects, ratio of fission to capture, etc., at energies where measurements are impossible at present. Much information can be obtained from low-energy resonances to help determine the distribution laws of level parameters, thus supplying information on which theories can be developed and nuclear models tested.

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Cooper Alloy 16″ stainless steel gate valve body illustrates the wide range of nuclear valves and fittings now being cast by Cooper Alloy for atomic energy service.

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Cooper Alloy Cast Stainless Steel
ASTM351-CF8 . . . . . 22-27,000 psi @ 650°F.(2)

(1) Figures represent minimum yield strengths required by ASME Unfired Pressure Vessel Code, Section VIII, and have been calculated as follows:

$$YS = \frac{\text{allowable stress}}{62.5\%}$$

(2) Figures taken from tests of actual heats of material manufactured by Cooper Alloy for the AEC's SPERT III program at Idaho Falls, Idaho (see Nucleonics, Aug. '57).

Strength: As the figures in the accompanying chart show, Cooper Alloy is currently achieving what many have described as "the impossible": producing 304 ELC stainless cast valves and fittings consistently with yield strengths of 22-27,000 psi at 650°F.! These superior physical properties are finding wide application in meeting the exacting strength and economy demands of the nuclear industry.

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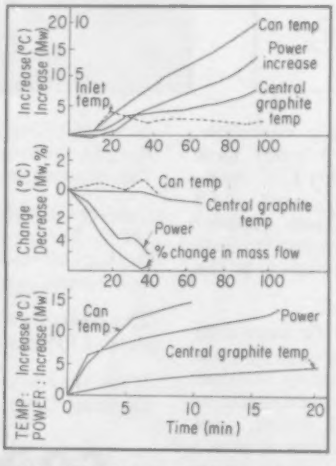
REACTOR PROJECTS

This article starts on page 72

flow can quickly affect the temperature of the gas returning to the reactor, and particularly good performance of the regulators is required during load changes and steady operation to avoid rapid temperature fluctuations.

Fuel elements. The fuel elements consist of natural-uranium rods contained in transversely finned Magnox cans about 100 cm long. There are six elements in a channel, stacked vertically, each bearing the weight of all elements above. The creep strength of uranium is low at operating temperatures, and, about the time the first reactor was being commissioned, experimental confirmation became available that, under irradiation, the zones of high internal stresses in the uranium set up by fission products considerably reduces the normal creep strength. To support the fuel elements so that they could withstand the desired irradiation period without excessive bowing, it was necessary to design an external brace that could be fitted to the existing element to support its central region. The brace consists of a spacer blade to the graphite channel wall that reduces the unsupported length of the element.

Each fuel loading consists of 10,000 fuel elements. In Reactor No. 1, only three fuel elements in the first, and one in the second, loading have failed. In the first loading of Reactor No. 2 there have been four positively identified burst elements.



CALDER CHART shows typical responses to changes in gas inlet temperature, coolant flow and control rod position after a fuel irradiation of ~440 Mwd/ton and for moderator temperature coefficient of + 4.7 × 10^{-5}/°C

Because of the difficulty of getting accurate temperature and channel flow measurements, it has not been possible to verify in detail the predicted heat transfer performance of the fuel elements in service.

Refueling equipment. Refueling is accomplished with the reactor shut down and depressurized but still containing CO_2. The equipment is mechanically complex, and initial rates of fuel handling were very low, mainly because of the necessity of relying on indirect indication circuits. Following installation of direct visual inspection arrangements, both by periscope and television equipment, fuel handling became much more satisfactory, and a reactor can now be completely refueled in about six weeks.

Coolant loops. Each of the four primary coolant loops is fitted with isolating valves in the 54-in. ductwork. These valves are not sufficiently leaktight to permit reactor operation to continue with an isolated loop depressurized.

In service, a coolant leak of about one tonne (2,204.6 lb) per day occurs. This does not appear to be from the primary coolant loop. Rigorous inspection has not revealed any porosity or leakage from the main ductwork or its welds, and the leak rate is considered to arise from small pipework and valves of the auxiliary loops. No accumulation of CO_2 arises in the reactor building, and the leak rate has presented no health hazards.

Loop shielding has proved adequate except at the blower control panel close to the main gas ducts near the circulators. Here the gamma radiation was 9 mr/hr, and, owing to the necessity for this area to be manned, light brick shielding walls had to be built around the main ductwork. The activity at the surface of the main gas ducts at the inlet to the heat exchangers is greater than about 40 mr/hr, but access to this area is not normally required. At the surface of the heat exchangers, the level varies from 10–4 mr/hr.

Very little particulate activity is carried around the loop by the coolant —less than 1 mc at any time—and only small amounts of radioactive solids are removed by the filters. After a period of operation of about a year, the heat exchangers were entered and inspected and the main circulators opened for inspection and maintenance. Inside the

advertisement

RARE EARTHS AS NUCLEAR POISONS

Cost and availability are no longer problems

a report by LINDSAY

Significant developments in the cost and availability of the neutron absorbing rare earths are redirecting the attention of control rod designers to the use of these materials as poisons in nuclear reactors.

Early interest in the rare earths was centered in europium because of its superb neutron absorbing properties and extended life. At that time, *all* the high cross section rare earths were scarce and expensive, and the metallurgical problems were relatively unexplored.

PRODUCTION FROM ION EXCHANGE

Since then, considerable experimental work has been devoted to development of practical fabricating techniques and the improvement of separation processes. Lindsay pioneered the first commercially installed ion exchange plant for the production of rare earth oxides in purities up to 99.99%.

Now, more than 100 columns with large production units of eighteen inch and sixty inch diameters are in continuous operation at the Lindsay plant. All the important neutron absorbing rare earths are now available, not only as the oxide which can be custom made with regard to density and particle size, but also as the boride, disilicide, or metal.

One important by-product of the advances in separation technology is that several of the rare earth absorbers have become common materials. The prices of oxides of gadolinium, samarium, erbium, dysprosium and europium have all tumbled sharply. Only europium is priced above $1 per gram; samarium and gadolinium are but a few cents per gram.

We know of course that europium is the most effective control rod material, but because of its scarcity, it will always be the most expensive rare earth absorber. Control rod designers now need to consider price, as well as properties, when selecting rare earth materials as control rod poisons.

Except in high flux reactors requiring few control rods, it is uneconomic to use europium as a primary poison; most commercial reactors do not need it except for its ability to absorb epithermal neutrons. Therefore, we suggest that control rod designers concerned with fluxes of about 2×10^{13} consider gadolinium as their primary poison, with samarium, dysprosium, erbium, or europium as additions to achieve desired rod worth. *Where both cost and performance are factors, the rare earths are no longer exotic, but actually superior to other materials.*

Furthermore, the prices quoted for certain mixtures of the rare earths may be well below those quoted for each material separately.

AMPLE CAPACITY TO MEET DEMAND

Lindsay has expanded production facilities and has ample capacity to meet anticipated demand for neutron poisons in the foreseeable future. Our technical staff is familiar with both control rod and rare earth separation technologies, and is available to help you select the most economical mixture of materials.

We have a wealth of technical data and ideas relating to the use of various rare earths in control rods, some of which is contained in our brochure "Rare Earths as Poisons in Nuclear Reactors." A copy is yours for the asking.

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Lindsay is the world's largest producer of *both* rare earth and thorium chemicals. Thorium and thorium oxides are being used and proposed more and more in new reactors in spite of the temptation to use uranium in order to capitalize on the temporary high buy back price for Pu. Thorium materials, particularly the metal, can operate at higher thermal conditions than their uranium counterparts, and show a somewhat higher thermal conductivity. The η for U^{233} is much higher at thermal neutron energies than η for any other fuel; the Th-U^{233} cycle is the best hope for breeding in thermal reactors.

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This article starts on page 72

heat exchangers there is a dust hazard, but surface contamination is <1,800 cpm/ft², and entry to the heat exchanger is possible for personnel wearing dust masks. The activity on the internal surfaces of the circulators is, in general, 6,000 cpm of combined beta and gamma activity so that no restrictions or special clothing are necessary for maintenance personnel.

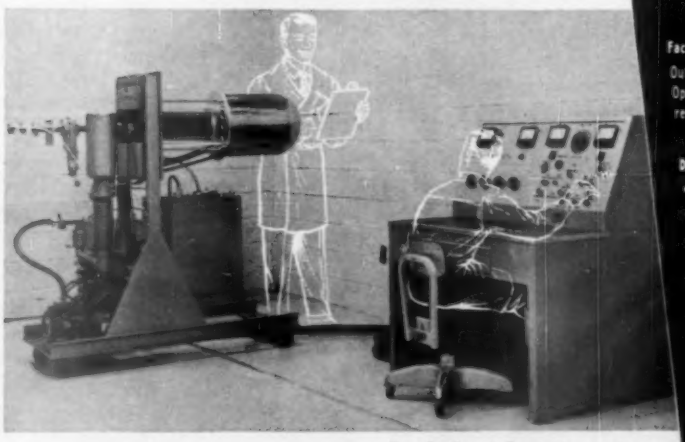
A-building

Britain

Four large two-reactor nuclear power stations for civilian electricity generation are daily rising higher into the skyline of rural Britain. When bids were sought in late 1955 by the nationalized electricity supply body, there was no limit on plant output: the only requirements were low specific capital cost and minimum unit generation cost. Metallurgical research on materials and fuel elements were assumed to be common to all competing designs and were undertaken jointly by the Atomic Energy Authority and the industrial groups. Private research by the groups —and in keenest competition—was directed to improving the design of individual components and to the further development of extended surfaces for heat transfer on the fuel elements and boiler tubes to secure the greatest possible increase in steam conditions without significant increase in operating temperature of the magnox fuel can.

Bradwell (*263*) and **Berkeley** (*264*) were the first two English stations awarded, and they are the furthest along, with the first reactor at each site due for startup late in 1960 and the entire station to be in operation at full power early in 1961. Both represent significant advances in output and steam conditions over Calder Hall (NU, Jan. '57, R5), yet were to be themselves eclipsed only eight months later by the plans for Hinkley Point (NU, Oct. '57, 24). Among the most significant innovations—aside from the use of 3-in. steel for pressure vessels, permitting higher operating pressures than the 2-in. steel of Calder—are the fuel element designs (see p. 102) and the machines for charging and discharging of fuel elements while the reactor is in operation. At Bradwell, for instance, the charge machine (see

New— a low cost neutron generator



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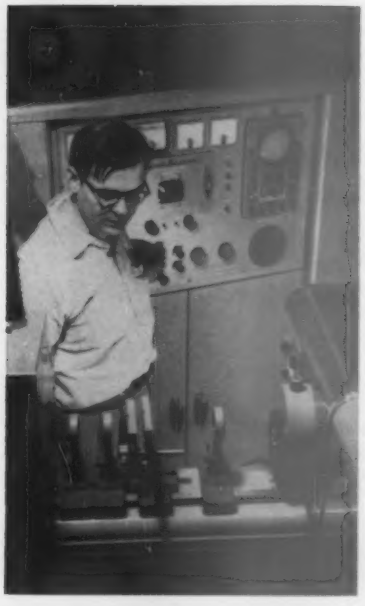

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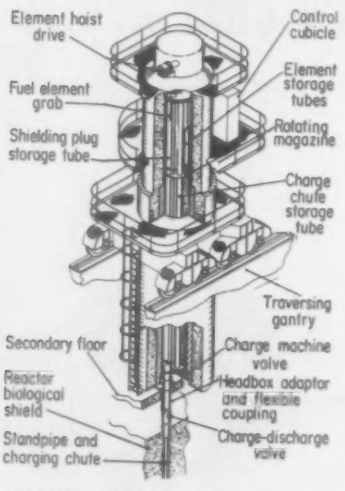

BRADWELL's fuel loading machine

above) is a heavily shielded tubular pressure vessel 55 ft high that can be accurately positioned over any of the 256 standpipes on the reactor vessel. Of these, 88 are refuelling standpipes each serving 34 fuel channels; the other 168 are for control rods, fission-product detection equipment and other mechanisms requiring to be inserted into the reactor. Inside the charge machine is space for new and spent fuel, a charging chute for insertion into the reactor vessel, and two standpipe shielding plugs, all handled by two internal hoists. Each refuelling standpipe is equipped with a power-operated valve at the top, below which is a removable shielding plug. The charge machine can be connected to any standpipe by removing a floor slab and fitting an adapter that takes up any errors in alignment between charge machine and standpipe. Once connected, the charge machine is remotely controlled from a room commanding a view of the entire reactor top; its hoist can remove the shield plug from the standpipe and insert the charging chute, giving the machine direct access to the core. Intense neutron and gamma radiation passes up the unplugged standpipe, but the machine's shielding reduces scattered radiation to <4 mrem/hr in the vicinity during operation. After the machine has recharged a channel, spent fuel is lowered from the machine through a discharge tube at one side of the reactor to an underground, water-filled storage pool for cooling.

The **Hunterston** station (*74*) being built in Scotland is much like Bradwell and Berkeley, awarded at the same time, but is the most powerful of the three at 320 Mw(e) compared with 300 and 275 respectively. Power costs at Hunterston are estimated at 4.7 mills/kwh for plant capital charges at 8%; 0.9 mills for fuel capital charges at 7%; fuel replacement costs of 1.0–1.6 mills, and 0.7 mills for operation and maintenance, for a 7.3–7.9-mill total.

Hinkley Point (*75*) with its 500-Mw(e) station output is the world's biggest nuclear power plant ordered so far. It is also expected to generate electricity for less than 7.6 mills/kwh, which is less than the national average cost of generation in Britain today. Yet the design is essentially similar to that of Calder and the three earlier civil stations, but with engineering parameters "notched up."

In a separate paper (*73*), the U. K. AEA discusses the analyses made of the industrial bids for the civil stations —assessments that occupied 100 senior professional staff for six weeks—to show that it was possible "to develop the Calder Hall design to produce economic nuclear power without departing radically from the technologies upon which the Calder Hall design was based." This paper also contains an appendix on conclusions of studies on CO_2 leakage from the pressure loops of the Calder reactors.

Basic design problems of the **Dounreay** 15-Mw(e) fast breeder are discussed (*274*) without reference to the fact that startup of this ambitious project has now been postponed a third time until mid-1959 (it was originally scheduled for April 1958) because of "diversion of effort necessitated by the Windscale accident, together with certain modifications to equipment within the biological shield." What the Geneva paper does reveal is that following the U. S. theoretical work consequent on the EBR-1 meltdown, "it became clear that with the proposed design of core assembly for Dounreay, at quite modest power ratings the radial temperature gradients in the core were sufficient to cause the fuel elements to bow toward the center; . . . the amount of bowing could be . . . appreciable." The solution chosen was to design "a form of tube nest which gave lateral restraint at the center but did not unduly restrict the cooling." Also the lattice was enlarged from 0.83 to 0.91 in. pitch and the number of fuel elements reduced.

(*Continued*)

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REACTOR PROJECTS

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France

The French are pursuing a very independent program, much more so than might be concluded from a very cursory glance that sees no more than the fact that they, like the British, are going the gas-cooled graphite-moderated route. The original single development program of the Commissariat à l'Energie Atomique, marked back in January 1956 by completion of France's first power reactor, G-1, is now being paralleled by the civilian-power program of the nationalized utility Electricité de France: each is now occupied with two large power reactors, the first of which, G-2, went critical in July (NU, Aug. '58, 19).

G-2 [G for graphite] and **G-3** (*1134*) are, like Calder Hall, intended primarily for plutonium production and only secondarily for power (40 Mw(e) each); but they have also served as developmental test-beds for new ideas (see Fig. 2). Their fuel channels are horizontal, not vertical. The reactor vessel is of prestressed concrete, 46 ft i.d. × 59 ft long, under 220 psi; this serves at once for pressure resistance and as biological shield. The coolant flow is divided into primary and secondary circuits: the primary in turn has two independent headers or manifolds, with one stream cooling the central core (581° F outlet, 1,806 lb/sec flow rate) the other a peripheral zone (669° F outlet, 387 lb/sec flow rate). The latter also cools the graphite wall and the thermal shield. The secondary circuit cools the concrete vessel. This arrangement of the coolant circuits makes for better utilization of the fuel.

There are 28 fuel rods 13 in. long in each of the 1,200 channels; the channels are 2¾ in. dia, the fuel rods 1⅛ and 1¼ in. dia with fins centering the cartridge in the channel. The natural-U cartridges are jacketed in magnesium alloyed with 0.8% zirconium; maximum surface temperature is 752° F.

Another unusual feature is the design of the apparatus for recharging while in operation (G-2 is the first reactor so equipped to be completed). At the loading face a shielded, pressurized lock chamber is moved into place; it can accommodate the 28-ft channel plug and enough slugs to refill a whole channel. The fresh elements are slid into the channel, pushing out irradiated

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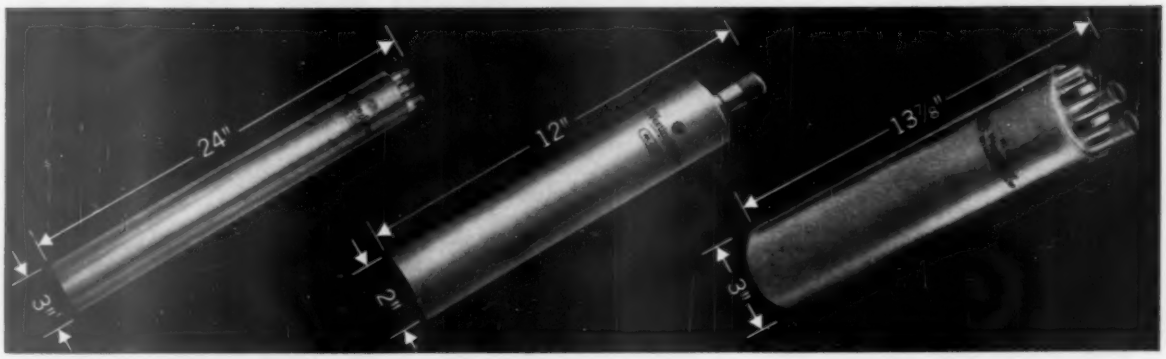
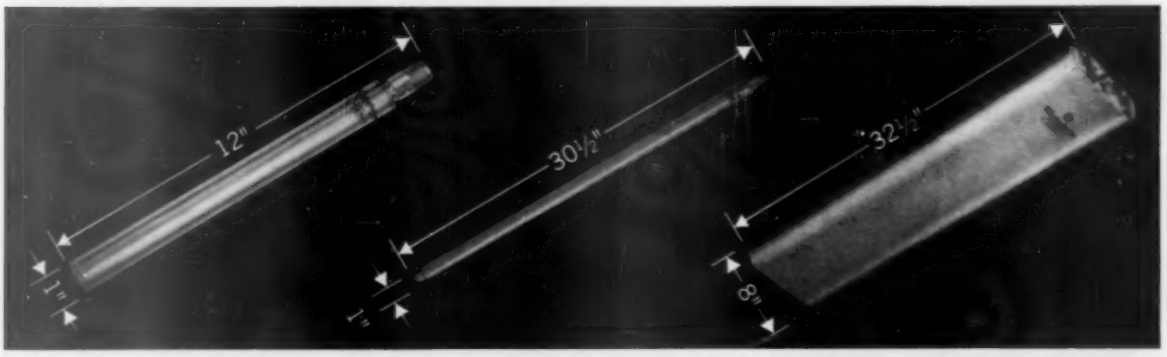


WL-6377 Compensated Ion Chamber:
Electrical compensation—guard ring design. All-magnesium construction—B^{10} coated electrodes. Thermal neutron sensitivity: 4.0×10^{-14} amps/nv. Gamma sensitivity: 3.0×10^{-11} amps/R/hour. Operating thermal neutron flux range: 2.5×10^{2} to 2.5×10^{10} nv. Passes Bu Ships 40T9 for vibration and MIL-S-901 for shock.

WL-6376 Dual-Range Fission Chamber:
All-aluminum construction—U-235 coated electrodes. *As a pulse counter:* Thermal neutron sensitivity, 0.7 counts/neutron/cm²; Operating neutron flux range, 2.5 to 2.5×10^{5} nv. *As an uncompensated ion chamber:* Thermal neutron sensitivity, 1.4×10^{-18} amps/nv; Gamma sensitivity, 4.2×10^{-11} amps/R/hour; Operating range, 5×10^{5} to 1×10^{10}nv. Passes MIL-STD-167 (Type 1) for vibration and MIL-S-901 for shock.

WL-6337 Uncompensated Ion Chamber:
Guard ring design. All-aluminum construction—B^{10} coated electrodes. Thermal neutron sensitivity: 4.4×10^{-14} amps/nv. Gamma sensitivity: 5.0×10^{-11} amps/R/hour. Operating range: 2.5×10^{4} to 2.5×10^{10}nv. Passes MIL-STD-167 (Type 1) for vibration and MIL-S-901 for shock.



WL-6367 BF_3 Proportional Counter:
All-aluminum body—enriched BF_3 filled
Thermal neutron sensitivity:
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Passes MIL-STD-167 (Type 1) for vibration and MIL-S-901 for shock

WL-6998 BF_3 Proportional Counter:
All-aluminum body—enriched BF_3 filled
Thermal neutron sensitivity:
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Operating range: 8×10^{-2} to 8×10^{3} nv
Passes MIL-STD-167 (Type 1) for vibration

WL-6938 High Sensitivity Proportional Counter: Contains four elements similar to WL-6998 in polyethylene moderator. Aluminum body
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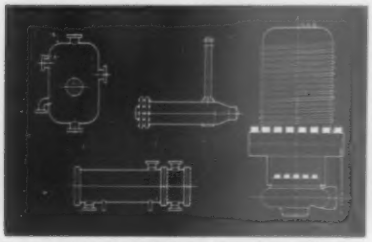

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REACTOR PROJECTS

This article starts on page 72

ones at the far end of the channel onto an inclined plane and a system of chutes. Since the cladding reacts with water, before the spent slugs can be put in the cooling tank they are first guided to a hot cell containing remotely-controlled machinery for welding the slugs into an aluminum tube. The burst-slug detection system resembles that of Calder Hall.

EDF-1 and **EDF-2** (*1135*) are now underway near Chinon on the Loire. The former, in an advanced stage of construction, is to be completed in spring; the latter is just starting. EDF-1 returns to a vertical channel alignment in a steel reactor vessel (33 ft i.d., 75 ft high), surrounded by a thick concrete shielding tower which in turn will be enclosed in a steel sphere. About 140 tons of natural U in 1,200 tons of graphite will be cooled by CO_2 at 368 psi entering the reactor at 284° F and leaving at 671° to produce 60 Mw(e). The 2-ft-long slugs are hollow: 1⅜ in. in dia, they have a 9/16-in. dia central hole, to minimize thermal gradients. Noteworthy is the main heat exchanger: it will be made up of 120 independent elements 2 ft dia and 94 ft high, weighing 20 tons. They will be separately removable for factory maintenance after decontamination without affecting the operation of the unit.

EDF-2, to be completed 18 months after EDF-1, will have a spherical reactor vessel packing 250 tons of uranium, do away with the containment sphere, and put out 170–210 Mw(e).

West Germany

The **pebble-bed reactor** (*1054*) started in July by Brown Boveri Co. and Friedrich Krupp for the AVR utility combine (NU, Aug. '58, 26) will operate with gas cooling within a temperature range of 1,112–1,832° F. This 15-Mw(e) pilot plant will be fueled with uranium carbide balls 2 in. dia and thorium carbide balls 2⅜ in. dia; the U is 20% enriched, for a total effective enrichment of 3–4% with an inventory of 50 kg U^{235} and 1,200 kg Th. The reactor, heat exchanger directly above and a tank for cooling down spent balls below all form one structural unit, so that the entire primary is shielded integrally. This simplifies construction, and affords safety since the primary circuit is activated by gaseous fission products diffusing from the fuel. Surrounding the core is a graphite reflector having four thick vertical fins protruding into the core; inside the fins are vertical holes from top to bottom to accommodate safety rods in the form of chains inserted from the top. Various gases are to be tested as coolants; early preference is—helium being unavailable in Germany—crude neon containing over 25% helium.

Belgium

Two papers both written jointly by Americans and Belgians describe the 11.5-Mw(e) pressurized-water **BR-3** (*416*) now being built at Mol for Belgium's Nuclear Study Center by Westinghouse, and a design for a 125-Mw(e) indirect-cycle **boiling-water reactor** (*1801*) planned "with a view to its construction in Belgium in the near future." Erection of the containment shell for the former is now underway and it is scheduled to go critical in December 1959. The latter, designed by W. H. Zinn's General Nuclear Engineering Corp., would have an oil-fired superheater; it "requires only a moderate investment and will result in a competitive kwh cost for Belgium."

Sweden

Adam, the 75-Mw(th) space-heat reactor (*136*) and **R-3**, a 36-Mw(e) power plus 55-Mw(th) space heat producer (*135*), are both under construction for startup in 1960–61 (NU, Dec. '57, 22). Both use natural uranium as oxide for fuel, heavy water for moderator and coolant.

Norway

The **Halden boiling-water reactor** for process steam (*559, 561*) is now on the way to completion by end of November, and critical testing is to start Dec. 1. This $3.7-million unit, which will supply steam for a paper-and-pulp mill, is located underground—as are also both Adam and R-3 in Sweden. The initial charge will consist of natural metallic U bars canned in aluminum, (supplied by Britain's AEA), spiked with enriched UO_2 pellets; the second charge will be uniformly slightly enriched.

Italy

Site selection and bid invitation procedure for the **SENN** power plant are described (*1364*). Nine firms in three countries (U. S., U. K., France) are now awaiting selection of one bidder.

Research Reactors

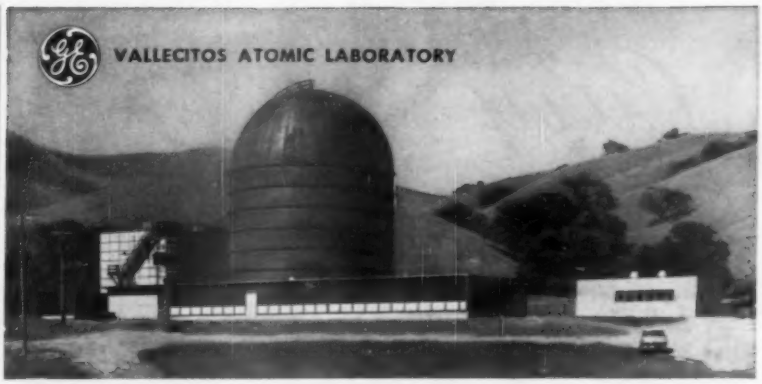
JEEP, the heavy-water research reactor of the Netherlands-Norwegian Joint Establishment for Nuclear Energy Research (JENER) at Kjeller, Norway, first went critical in July 1951, and was reviewed in Geneva-'55 paper 888. Developments since 1955 are summarized below (*1475*).

The impurity content of the heavy water has increased over the years, causing an increased D_2 content in the reactor atmosphere. To deal with this, the copper catalyst in the recombination system has been replaced by a better one of platinum deposited on aluminum oxide. Further accumulation of impurities will be prevented by using a purification unit, just completed, consisting of a ceramic filter followed by anion and cation exchangers; part of the D_2O flow circuit will be channeled through this unit.

The reactor was originally designed to operate at 100 kw. The present cooling system could accommodate 800 kw, but radiation levels in the reactor building would become so high at that level as to require additional shielding, assuming that fuel-element surface temperatures did not exceed a safe value. The level of 450 kw has therefore been adopted as standard. The degree to which the reactor is used has also increased over the years: it is now operated around-the-clock, weekends included.

All the fuel elements in the reactor are from the original charge, and most have been in for the life of the reactor. They are inspected twice a year. Ten days is accepted as the minimum cooling period for inspection; rod activity is then about 4,000 c. The formation of blisters on the fuel elements, reported in 1955, now seems to have stopped and no further growth of those present has been observed. The blisters are believed to have been caused by moisture between fuel and can, not by irradiation. The elements are covered with a layer of aluminum oxide that has not altered. The central elements have now accumulated 250 Mwd/ton and will be replaced in the near future.

Armour Research Reactor (ARR) went critical June 24, 1956 (*418*), and as with any new reactor, a number of operational difficulties were uncovered. After 26 kwh of operation, recombina-



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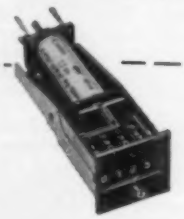

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REACTOR PROJECTS

This article starts on page 72

tion of radiolyzed hydrogen and oxygen was detected in a region immediately over the surface of the fuel solution. It became clear that ignition of the hydrogen and oxygen mixture was possible because of the specific design of the gas-flow system and that the recombination was being triggered by small fragments of catalyst that had been carried up into the core region. Preparations were therefore made to remove the original core and replace it with one incorporating a slightly redesigned gas-flow pattern.

Decontamination was accomplished by flushing with a one-to-one nitric acid solution and distilled water. Strong decontaminating solutions could be used because the primary system is fabricated entirely of 18-8 stainless steel; gold gaskets are used for sealing the two stainless steel pressure transducers, and stainless steel O-rings are used on the recombiner pump.

After replacing the core, the reactor operated routinely at powers up to 10 kw for about six months after which a fuel sample was taken to ascertain whether any appreciable corrosion had occurred. No indication of corrosion was present, but five days later the reactor became subcritical. At first it was thought that precipitation of the fuel was the cause; after several weeks it was shown that the cause of the difficulty was in fact incomplete closure of valves manipulated for the sample-taking. The valves did not seat properly because a small amount of catalyst dust had deposited on the valve seats during sampling. This allowed fuel to leak out of the reactor core into a fuel drain tank in the subpile room.

When operation was resumed, and while proceeding to a 10-kw power level on a 15-sec period, the catalyst-bed heater fuse failed, allowing the build-up of hydrogen and oxygen because the catalyst was cold and hence inactive. The next day the catalyst-bed heater burned out. To replace the heater, located in the recombiner and therefore in a high gamma-ray field, two men were sent into the secondary enclosure for the one minute it required to do the job; they absorbed 150 and 90 mr.

After the heater replacement it was noted that various pressure drops and flow-rates in the system were slightly changed from normal. Similarly, a

change in catalyst-bed temperatures and a slight increase in reactivity amounting to 0.15% was observed. From this information it was concluded that the fuse failure had caused ignition of hydrogen and oxygen which, in turn, forced some water out of the level-control weir in the recombiner. A valve was manipulated to refill the weir.

The reactor was again placed in operation and, about two weeks later, gas phase activity to the extent of 3×10^{-6} $\mu c/cm^3$ was detected. As a result of a series of wet swipes taken on components of the gas recombination system, activity was detected at a particular weld on the aspirator pump. The reactor was decontaminated, the weld repaired and the entire system leak-checked. Fuel was returned to the core and preparations made to resume operations.

Four days later, it was determined that the pump pressure transducer had failed; once again the fuel had to be dumped. It was decided to relocate the pump transducer, system-pressure transducer and gas-flow measuring assembly behind shielding so that any of these units might be replaced without decontaminating the system.

Since Nov. 25, 1957, the reactor has been in routine operation and no further malfunction of any sort has been encountered. During the activities that brought the reactor to its present trouble-free status, a number of facts have become clear:

- The closed recombination system has proven to be remarkably versatile, and adequate provision has been made to permit the reactor to recover from almost any abnormal condition.
- The primary system is capable of containing hydrogen and oxygen ignitions without damage to the components.
- The system can withstand decontamination procedures without injury.
- Quite complicated manipulations with large quantities of radioactive gas and liquid can be done safely. No overexposures or accidental release of activity were encountered.

A considerable degree of confidence has developed during the course of these early difficulties. The reactor system seems to have all the design features necessary to recover safely from almost any form of equipment malfunction or operator error.

Brazilian pool. As there is no published literature on behavior of a swimming-pool reactor at powers

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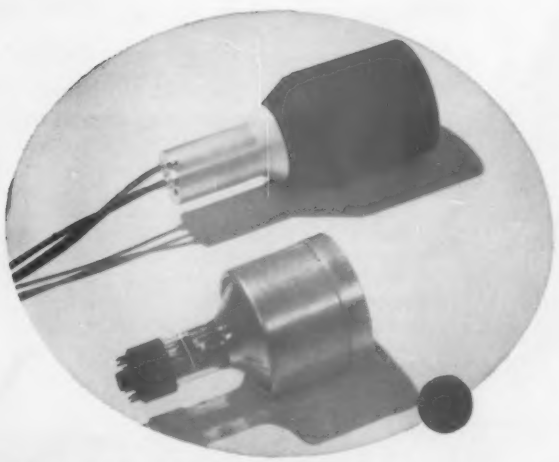

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This article starts on page 72

higher than 1 Mw, the first results obtained with the IEAR-1 are noteworthy (*2279*).

In experiments at power levels higher than 100 kw, forced cooling of the core must be used; an interlock system in the log-n recorder is provided to scram the reactor if the primary coolant pump fails. In these experiments, a flow of 2,200 gpm in the primary system and a flow in the secondary of 1,500 gpm are used because a high rate of vibration in the heat exchanger was found to occur at flow-rates higher than 1,600 gpm in the secondary.

As power is gradually raised, special attention is paid to the temperature of the water entering and leaving the core, and when a power of 1 Mw is reached, the reactor is kept at that level for 5 hr. In the experiments made at powers of 2.5 and 5 Mw, the linear level channel required special attention to permit detection of any effect that might be caused by possible bubble formation on the surface of the fuel element plating. However, as predicted by calculations, no such effect was observed and the reactor operated continuously for hours at 5 Mw. Measurement of the radiation levels on the pool surface and in the beam hole doors, and of the activities in air and water, have shown that operation at such a high power level is possible. A detailed report on the observed activities shows that the high level of radiation observed in the air is caused by a large amount of turbulence in the pool surface. This prevents the formation of a clean hot-water cushion on the surface of the pool that would normally prevent hot atoms from coming to the surface and diffusing into the air. Because such turbulence is caused by the faulty design of the primary return water outlet, a new design will be tried.

New Designs

With some 81 known research reactors in operation in the world, designs are becoming more sophisticated. Thus the West Germans, after having bought four U. S. and two British reactors, are building a high-flux research reactor of their own design, FR-2 (*1630*), as the first unit of a major nuclear research center at Karlsruhe. Natural-U fueled and D_2O moderated, it is similar to Canada's NRX, but uses D_2O also for

cooling. Maximum thermal flux is 3×10^{13}. Fuel elements will have an aluminum tube soldered around the Al-canned U rod to carry coolant. Construction was started last year, and criticality is slated for late 1959. Although producing 3 kg/yr plutonium and some U^{233}, its main purpose is reactor and fuel element development.

France completed in July her third major research reactor, EL-3 (*335*), with a thermal flux of 10^{14}. Also D_2O moderated and cooled, its fuel is 1.4% enriched U obtained from Britain. The 13-Mw(th) unit has some 58 horizontal and vertical beam holes besides a thermal column and other experimental facilities. It has been at full power since December 1957. France has also designed two pools (*1205*) dissimilar in design and function although both obtaining 10^{13} flux at 1 Mw(th) using Babcock & Wilcox plate-type elements. One, Mélusine, is designed for maximum flexibility as a research and training tool; it went critical in July. The other, Triton, is specifically adapted for shielding studies. Both contain a number of original devices.

Italy's first research reactor is a CP-5-type based on the modified design used in the MIT reactor at Cambridge.

Italy's first research reactor, now nearing completion at Ispra 40 mi N of Milan (*1383*), is being supplied by ACF Industries; it is a CP-5-type based on the modified design of the MIT reactor at Cambridge (*417*) which went critical in July. Ispra will operate at 5 Mw(th) and yield 8×10^{13} thermal flux.

Britain's Associated Electrical Industries Ltd., a partner in the A.E.I.–John Thompson consortium, built England's first private reactor—a light-water cooled and moderated, highly-enriched U unit called Merlin (*313*). An adaptation has been built for the Atomic Weapons Research Establishment (*1464*). Power is 5 Mw(th), thermal flux 5×10^{13}.

Aerojet-General Nucleonics described its AGN-211, a modification of its solid-homogeneous polyethylene-UO_2 concept, made familiar in the AGN 201, to a pool-type (*2420*). Also Australia discussed the design of facilities associated with its research reactor HIFAR (*1093*); Canada told about its big new 3×10^{14} research reactor NRU (*211*), India its first reactor, the pool Apsara (*1625*), and the two countries collaborated on a paper on the Canada-India Reactor (*1704*), a modified NRX that Canada is building for India.

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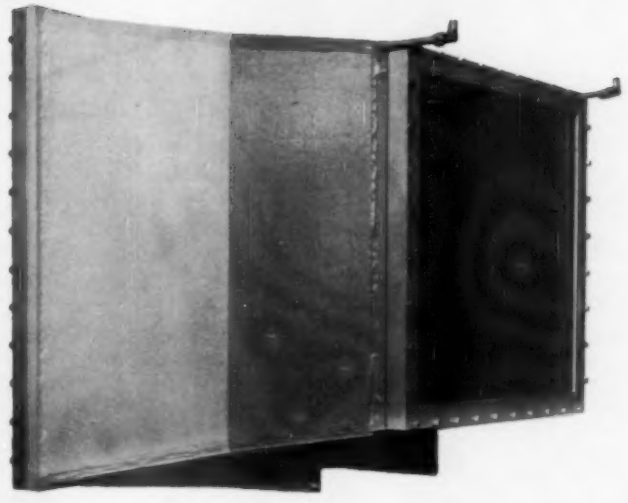

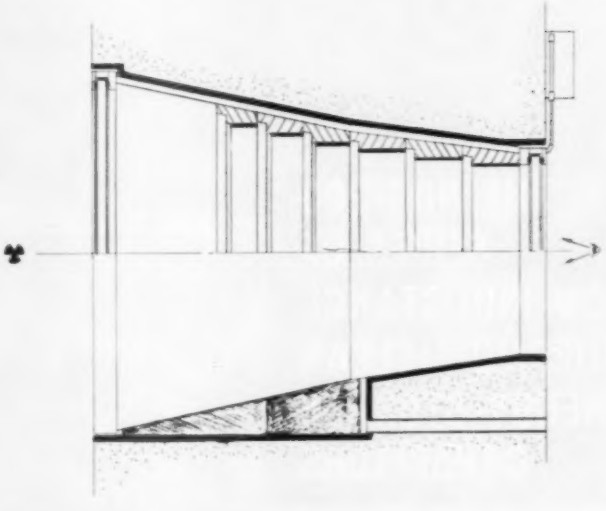

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FUSION

This article starts on page 66

vices depending on self magnetic field being exploited at Berkeley and Livermore (*1064*). Even the "straightforward" linear pinch is being taught new tricks as the axial magnetic field is programmed in time coincident with the pinch that sweeps it toward the axis so as to provide more stable field configurations (*369*).

Rotating plasmas are provided by the Homopolar device (Fig. 4b, p. 67), which uses radial electric and longitudinal magnetic fields to exert torque on the plasma much as in a homopolar disk-type motor. The rotational kinetic energy can later be converted to plasma heating by viscosity effects, thus providing a heating mechanism more powerful than simple ohmic heating, which becomes less efficient as the temperature rises and plasma resistivity drops. An important incidental use of Homopolar is as a fast discharge capacitor: dielectric constants of 10^4–10^7 and current capabilities of 300 kiloamp at rates of 5×10^{11} amp/sec have been produced easily. (See also Los Alamos' Ixion, p. 69.)

Triax. Theory indicates that a current-carrying plasma sheet of infinite extent has at least neutral stability (like a ball on a table). The cylindrical analog of this has been provided in Triax (*2349*), the triaxial pinch, where a "sleeve" of plasma is pinched between coaxial conducting cylinders (see Fig. 4c, p. 67). Neutrons have been produced with Triax, but there is no knowing yet if they are truly thermonuclear in origin.

Screw-dynamic pinches. By applying a helical twist to the return conductor surrounding a pinch tube simultaneous longitudinal and axiumthal magnetic fields can be established within (*372*). This configuration should provide greater stability than the simple pinch, but experiments show that it too is unstable. The associated neutron emission is not reduced by the introduction of impurities, which suggests that their origin might nonetheless be thermonuclear —a larger device now being built.

Christofilos' Astron

The Astron, one of the most unusual of the Sherwood approaches, is still in the planning stage (*2446*). Conceived by N. C. Christofilos of strong-focusing fame, the idea is to use a rotating cylindrical beam of electrons of relativistic energy to provide a confining magnetic field configuration such that the lines close on themselves within a vacuum vessel. The electron beam also serves to ionize and heat the plasma. Experimental investigations involving a pulsed 100-amp 1-Mev beam are now underway at Livermore—the aim: to study the magnetic field configurations that can be produced. In spite of its apparent complexity, the Astron offers sufficient potential in terms of stability that it is judged well worth pursuing.

In one of the few references to costs in the Sherwood papers it is stated that "it appears feasible to build and operate an Astron power reactor competitively with conventional power plants . . . the particular items associated with the injection and acceleration of electrons constitute less than 15% of the total cost . . . the balance involves familiar equipment."

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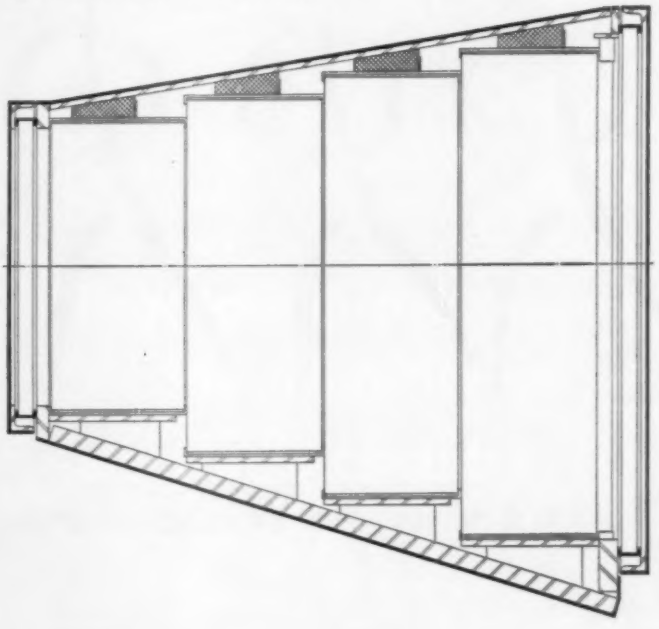

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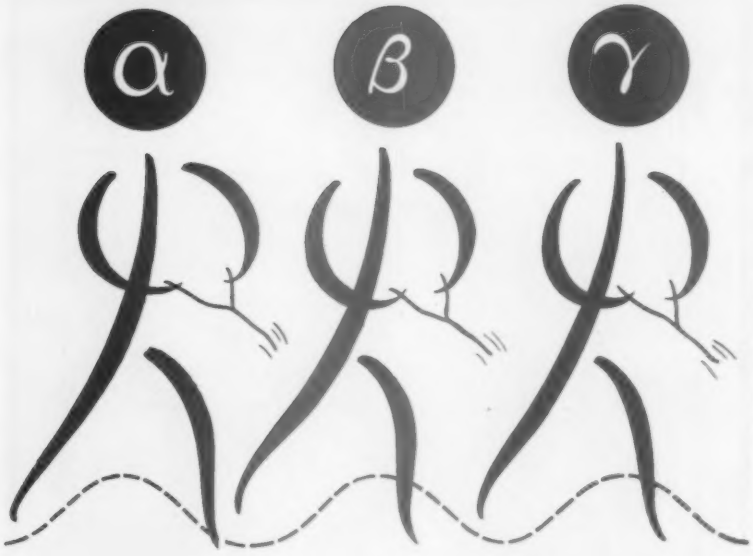

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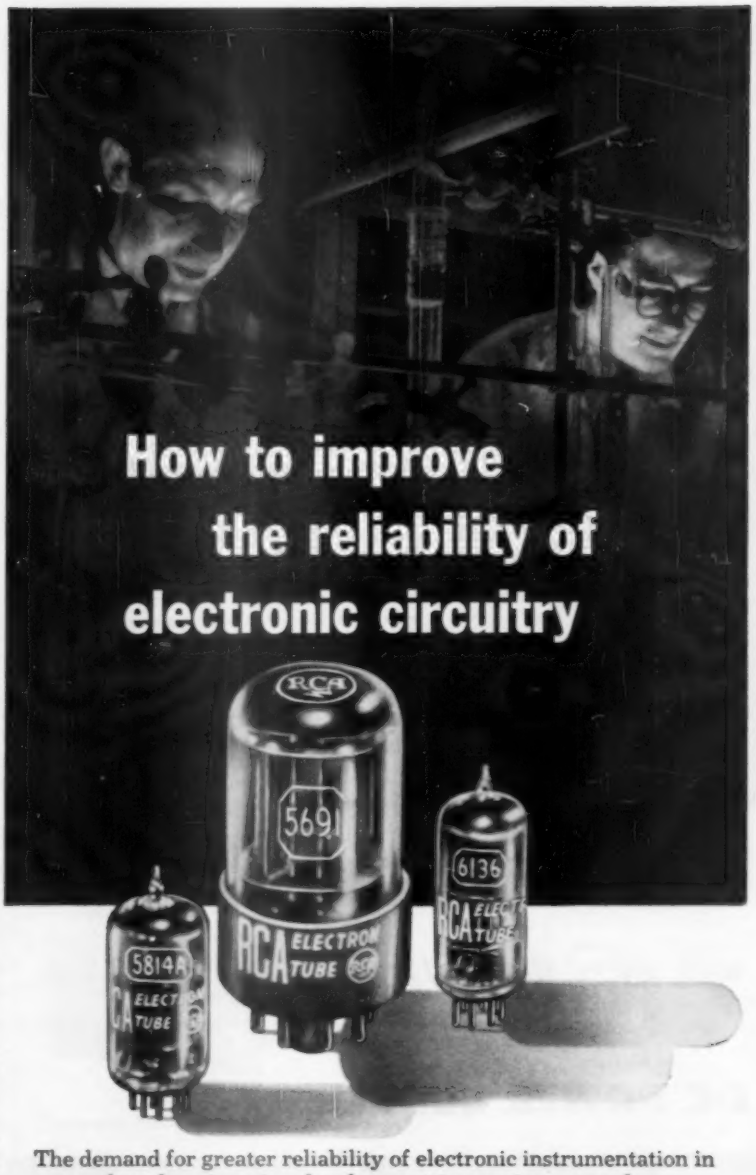

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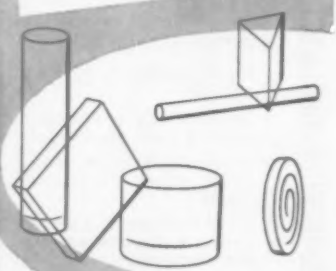

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Late NEWS

The entire Roundup section in this issue is devoted to the Geneva Conference. Other news developments are reported here in the next few pages.

▶**A U. S.-Euratom reactor program,** imperiled to the last by the unrelated but torrid hassle over funds for the U. S. atomic energy program, won the unanimous endorsement of the Joint Committee on Atomic Energy last month then cleared the House and Senate less than eight months after Euratom came into being on January 1.

The Congressional sanction, though modifying the program, signalled the start of U. S.-Euratom talks this month on administration of the program and an early beginning of negotiations between U. S. suppliers and Euratom utility companies on the reactors to be built under the 1-million kw(e) program.

JCAE, disgruntled by the late submission of the proposal and bristling at President Eisenhower's declaration that he will hold back JCAE-sponsored funds for acceleration of the U. S. nuclear program, forced the Administration to accept some changes before agreeing to clear the arrangement:

1. The 1963 target date has been extended to 1965 for two of the six reactors contemplated; this provides some leeway for concepts other than light water to qualify.

2. JCAE will review all contracts under which U. S. firms sell fuel elements to Euratom customers under AEC guarantees; the ceiling for U. S. payments under the guaranty program was set at $90-million.

3. A joint research and development program was approved but JCAE authorized only $3-million for the first-year U. S. share; the agreement calls for a $50-million contribution by the U. S. over the next five years.

Euratom had been awaiting this U. S. action before entering into negotiations with the U. K. for a comparable program encouraging gas-cooled reactor construction; it has generally been assumed that about half of the early nuclear power capacity in the Euratom countries would be of the Calder-Hall type.

▶**Setting the stage** for adoption of a new plutonium buyback policy next year, AEC and the Joint Committee on Atomic Energy agreed shortly before the adjournment of Congress to abandon the present minimum guarantee of $30/gr for military-grade Pu when it

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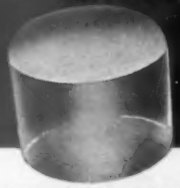

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expires June 30, 1963. Capping several months of negotiations, AEC and JCAE agreed on a ceiling of $16.12/gm for weapons Pu purchased from domestic reactor operators through June 30, 1968; the fuel-grade price would remain at $12/gm ceiling and both prices would be offered to foreign reactor operators under contracts up to ten years. Legislative action embodying the new policy was postponed until 1959 because agreement on it came too late in the 1958 session.

▶**The Burlington (Vt.) Light Dept.** is considering construction of a 50-Mw(e) power reactor, will make a final decision "in a few weeks." If the decision is "yes" Westinghouse will handle development, design and manufacture with J. F. Pritchard and Co., Kansas City, Mo., as architect-engineer. William A. Stebbins, superintendent, said the department was not limiting its studies to light-water reactors.

▶**The Tennessee Valley Authority's** technical staff, NUCLEONICS learned, is conducting an investigative study of two gas-cooled reactor concepts: 1. the GCR-2 which TVA helped Oak Ridge National Laboratory to design earlier this year (NU, July '58, 118); and 2. the pebble-bed reactor (NU, July '58, 24) under design by Sanderson & Porter. Both types promise to approach modern steam conditions—1000° F. and up—and appeal to TVA as those with the most economic promise. A staff recommendation on prototype construction of either concept will go to the TVA board of directors "in 2–3 months."

▶**The atomic energy industry** lost one of its most respected leaders when **Gordon Dean** was killed in an airplane crash on Nantucket Aug. 16. Dean, second chairman of AEC, was named to the Commission early in 1949 and served as its head July 11, 1950–July 1, 1953. He then joined General Dynamics as senior vice president for nuclear energy, where he had much to do with planning for General Atomic division's $10-million Hopkins Laboratory of Pure and Applied Science in San Diego, and with coordinating GD's many nuclear activities.

▶**Ralph J. Cordiner,** chairman of General Electric, has called off a proposed meeting of reactor manufacturers and utility firms at Woodside, Calif., Sept. 27–28, to discuss the nation's atomic energy policy (NU, July '58, 22). In-

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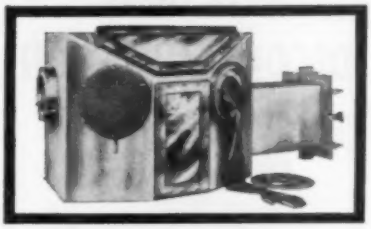

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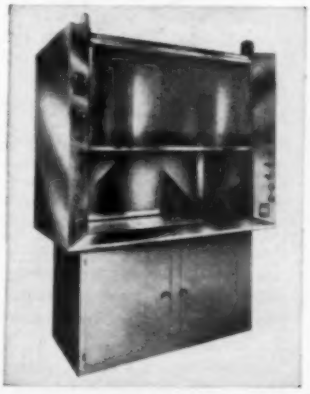

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Late NEWS

This article starts on page 152

stead, atomic energy will be added to the agenda of a GE-sponsored utility industry meeting earlier that week at Woodside. GE spokesmen denied that the plan for a special nuclear gathering had been dropped because of insufficient interest but no other reason was offered; no reactor manufacturer other than GE will attend the utility meeting.

▶**The United Kingdom Atomic Energy Authority** has concluded a five-year agreement with Power Reactor Development Co., Detroit, for the exchange of information on fast reactor development (effective Oct. 1).

▶**Government and industry** interests in Israel have joined forces to establish Isratom, a nuclear engineering firm, to direct all Israeli nuclear projects, including power reactors for stationary and marine use. Delek, the government-owned fuel company, will control 75 per cent of the stock.

▶**Argonne Lower Power Reactor** (ALPR) went critical at 10:30 pm, August 11, at the National Reactor Testing Station, Idaho. A prototype of power stations for remote site operation, ALPR will be run up to full power—3,000 Mw(th)—only after preliminary tests.

▶**AEC has raised** the limitation on the amount of plutonium it will ship to any given country—from 10 gm to 260 gm. The increase will permit wider construction of subcritical assemblies using Pu as a neutron source (80 gm per assembly).

▶**AEC has launched** a broad, first-phase effort in its new isotope development program without waiting for Congress to appropriate $5-million requested by AEC to support the program until June 30, 1959 (the House cut $3-million from the request—NU, July '58, 17—but AEC expected at least partial restoration in the Senate). Steps taken while AEC awaits final action on the request include: 1. the award of seven study contracts on the potential of new or existing isotope applications; 2. the award of four contracts in the high level radiation field; 3. development of a special isotope training course; 4. a study contract with the National Industrial Confer-

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ence Board on the economics of radioisotopes (preliminary report due this month); and 5. a contract with Stanford Research Institute on the medical potential of isotopes.

The seven firms awarded isotope study contracts are: **Armour Research Foundation,** development and production of isotopic X-ray sources from the secondary radiation of beta emitting isotopes in the region 5–200 kev; **Tracerlab,** development of a radiochemical analyzer for nonradioactive gases; **General Atomic,** a study of the Szilard-Chalmers process for the production of short-lived, high specific radioisotopes; **Radiation Applications,** study of the technology and application of large fission product beta sources: **Battelle Memorial Institute,** development of radioactive tracer quality control systems; **Nuclear-Chicago,** exploration of radiometric methods of industrial processes and controls; **Nuclear Science and Engineering,** development of devices and techniques for industrial applications of radioisotopes.

Pending the outcome of its appropriations fight, AEC plans to proceed to the second phase early this fall—publication of a program brochure and an invitation to private industry to bid on a series of contracts looking to development of further industrial applications of radiation. Construction of a high level irradiator is scheduled by AEC late this year or early next for process irradiation by both AEC and industry.

▶**Congress has restored** the first-year construction funds ($20-million) for this country's first commercial-type, gas-cooled reactor; the House had earlier refused to vote the funds (NU, Aug. '58, 17). AEC is now free to proceed with a survey of industry interest in sponsoring the reactor and to go ahead on its own if no industry proposal is made.

▶**The long-range reactor program** informally approved early this summer by AEC and the Joint Committee on Atomic Energy (NU, Aug. '58, 18) will be circulated to industry this month for comment, according to Congressional sources.

▶**AEC's Advisory Committee on Reactor Safeguards** found last month that General Electric's test reactor (GETR) "may be operated . . . without undue hazard to the public"; AEC then called a Sept. 17 hearing on issuance of an operator's license. ACRS took the oc-

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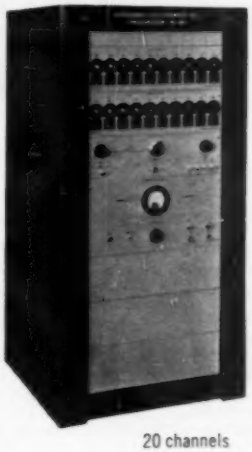

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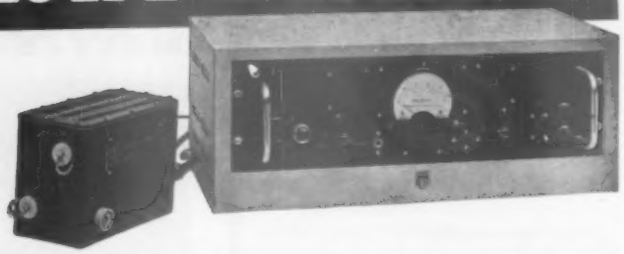

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Late NEWS

This article starts on page 152

casion to stress the need for permitting test-reactor operators—with their uncertain operating plans—more freedom of action under licensing arrangements than other reactor operators.

▶**President Eisenhower** last month vetoed the bill authorizing a nuclear-powered ice-breaker for the Coast Guard for a reason which has become a prime bugaboo for the U. S. atomic energy program: the mounting federal budget deficit. "A continued disregard of our budgetary problems through the institution of unneeded new programs and projects can only add to inflationary pressures to the detriment of all the people," he said in his veto message. The icebreaker joins a growing number of AEC- and Congressionally-sponsored nuclear projects declared "unneeded" by the White House, including a plutonium-production reactor, a nuclear tanker and a long list of research projects.

▶**AEC has awarded** Nuclear Development Corp. of America a second contract under the preliminary design and development phase of NDA's joint project with Chugach Electric System to build a sodium-cooled, D_2O-moderated reactor near Anchorage, Alaska. The contract adds $2.3-million to the $1.7-million already awarded NDA for preliminary work; among other things, it will support modification of reactor design to permit construction of a pilot plant.

▶**AEC's distribution policy** for heavy water was liberalized last month to give the domestic and foreign operators of research, medical and test reactors a choice of lease or sale. Lease price is 4% per annum of the sales price ($28/lb at present). Quantities available for lease range from 2,000/lb minimum to the initial inventory requirement of a given reactor.

▶**The second in a series** of test shots under Project Plowshare—the peaceful application of underground nuclear explosions—is planned by AEC next summer in a salt bed 25 mi southeast of Carlsbad, N. M. Bomb yield would be 10 kilotons, compared with the 1.7-kiloton yield of Rainier, first underground shot (NU, Apr. '58, 21). Principal function of the test will be to pro-

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vide data on production of radioisotopes and seismic effects; AEC is considering further tests (including one in Alaska in 1960—NU, July '58, 27) to prove out bomb explosions for mining, excavations, oil shale production and other uses.

▶**AEC has established** a one-man Office of Hearing Examiner—Samuel W. Jensch is the man—to conduct the Commission's regulatory hearings. The office will be directly responsible to the commissioners.

▶**The $500-million federal** indemnity protection of the Price-Anderson bill has been extended to the NS *Savannah*, now under construction. Congress has also agreed to indemnify reactors operated by nonprofit educational institutions, provided they make their own provision for coverage up to $250,000.

▶**A group of seven public power** companies in northwest Germany (SKW) plans construction of a 100–150 Mw(e) nuclear power plant in Lower Saxony for completion in 1964. SKW is reportedly talking with both U. K. and U. S. firms. A group in southwest Germany—AKS (Stuttgart)—is reported shifting its interest from a Calder-Hall type to an organic-moderated reactor of U. S. design. A decision to order a 100 Mw(e) plant may be made this year.

▶**Four nuclear submarines** passed milestones last month while *Nautilus* and *Skate* were garnering headlines for their epic polar trips. *Seadragon* was launched at Portsmouth Navy Yard Aug. 16, *Triton* at Electric Boat Aug. 19. *Sargo*, first West Coast-built A-sub, went to sea Aug. 3 to begin her builder's trials, *Swordfish* followed on her maiden voyage Aug. 9. The last two, sisters of *Skate*, are expected to be turned over to the fleet in about a month to become the fourth and fifth nuclear subs in commission.

▶**Frank Pittman,** AEC's industrial development chief, has taken on the acting directorship, division of reactor development. AEC is still hunting a successor to W. Kenneth Davis (NU, July '58, 21).

▶**AEC's Materials Testing Reactor,** Arco, went critical on its plutonium fuel core (NU, Aug. '58, 24) early in August, reached a power level of 5,000 Kw(th) later in the month, and was expected to reach 30,000 Kw by Aug. 31.

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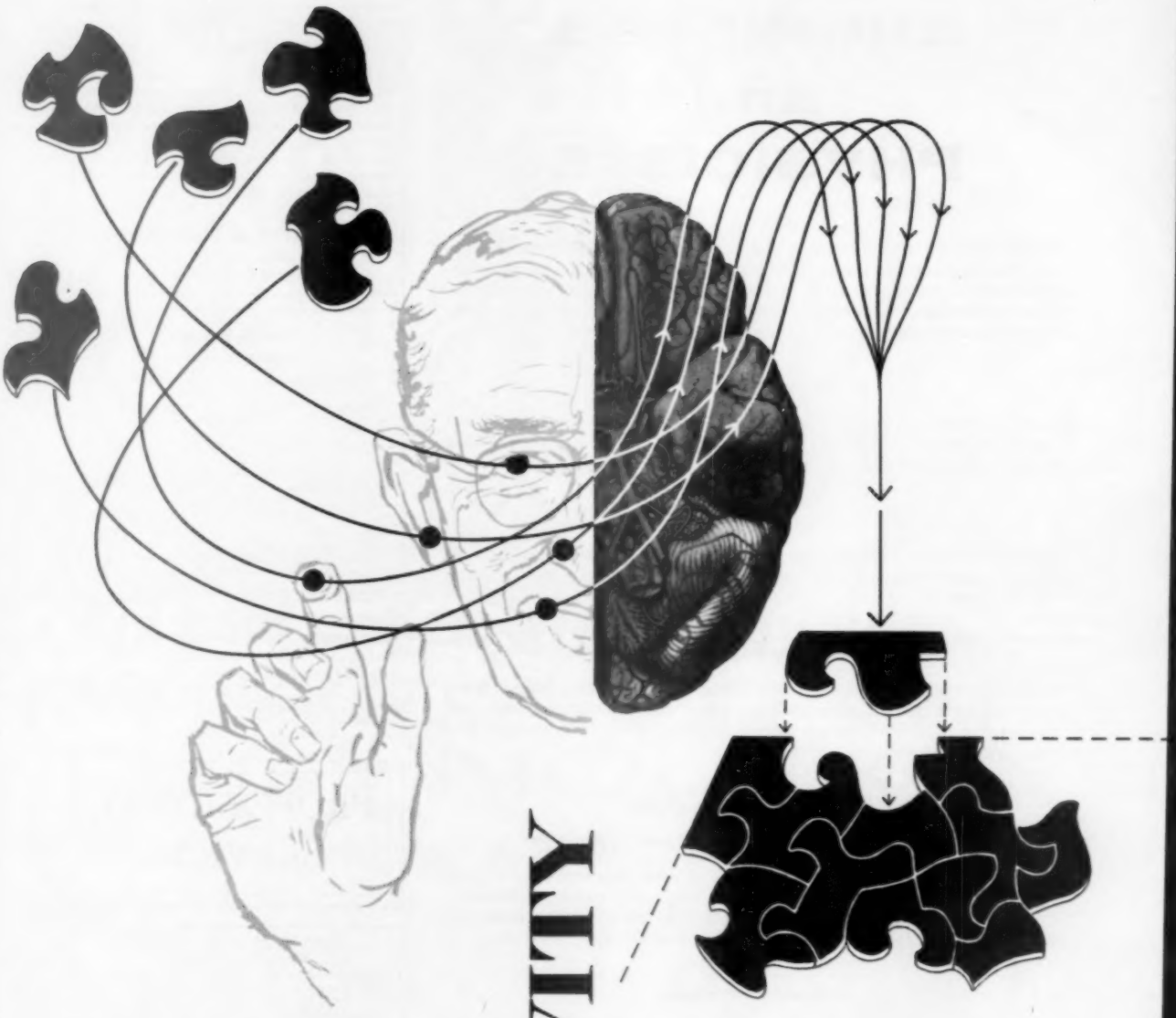

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Richard L. Mehan is one of several metallurgists working on the development of new and more reliable reactor materials at KAPL. Dick Mehan joined General Electric after graduation from MIT in 1950 and came to KAPL in 1953, following two years of army service. He is also studying advanced metallurgy, evenings at Rensselaer Polytechnic Institute under G.E.'s advanced degree program.

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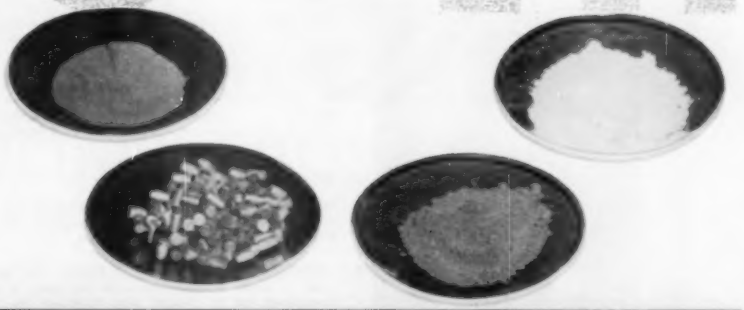

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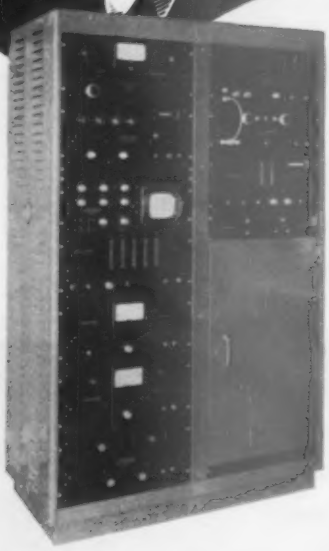

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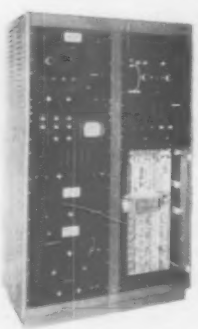

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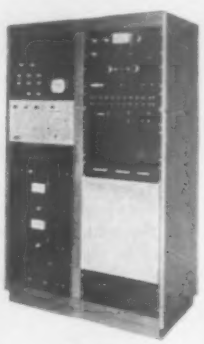

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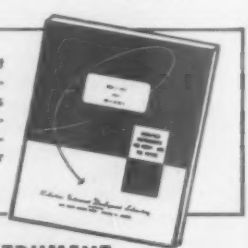

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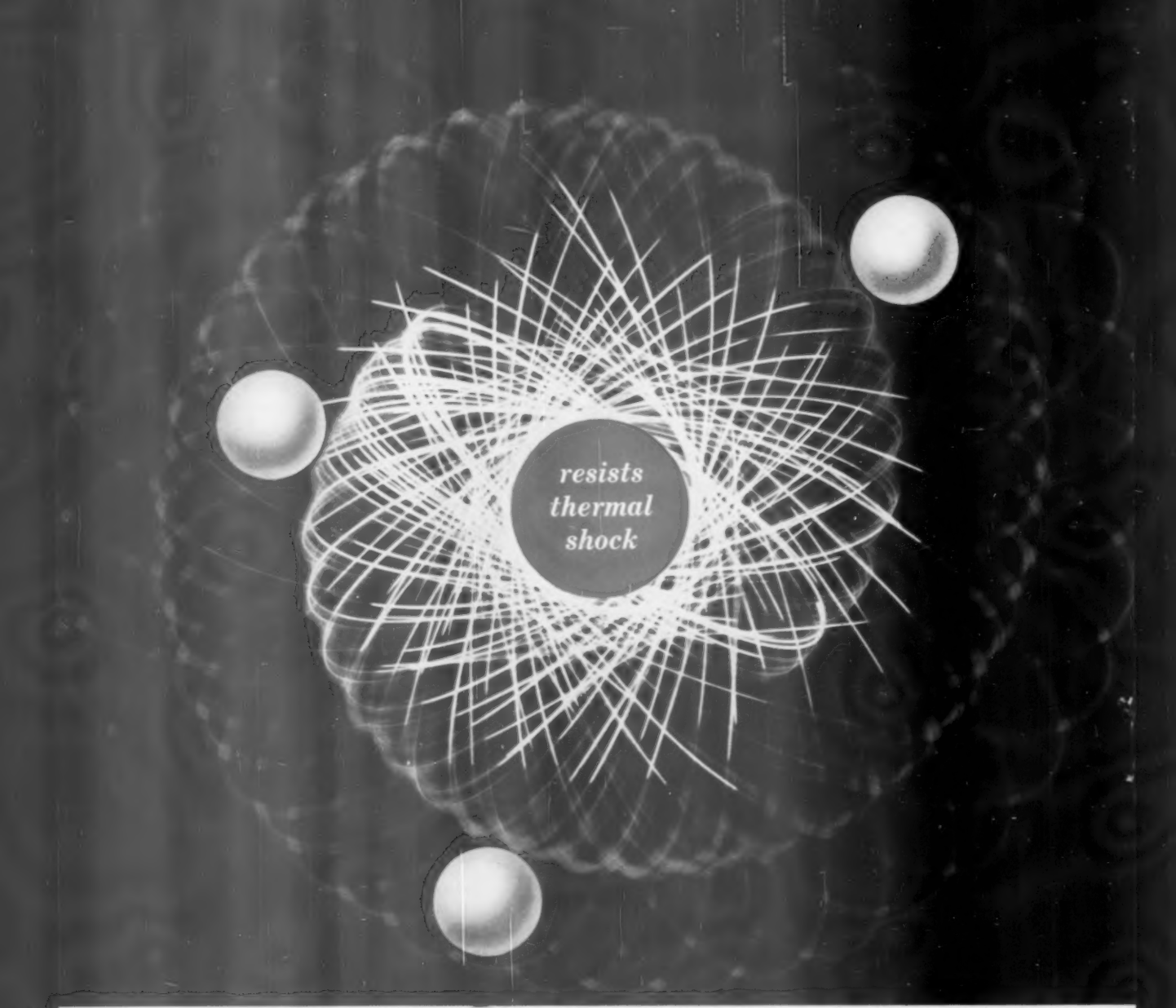

A photographic light pattern interpretation of graphite's thermal shock properties.

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